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Accident Assessments for Idaho National Engineering Laboratory Facilities

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ABSTRACT

These accident assessments were performed in support of the *Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Environmental Impact Statement.* Accidents related to Idaho National Engineering Laboratory (INEL) facilities involved in the following program areas were analyzed: (1) spent nuclear fuel management, (2) waste management, (3) environmental restoration, and (4) decontamination and decommissioning. Existing INEL safety analysis documentation, and discussions with and reviews by INEL personnel, were used to determine accidents at the INEL that could have the potential for producing adverse impacts to the health and safety of workers and the public. The potential consequences of accidents were based on existing INEL safety analysis documentation where available; where existing analyses were not available, independent accident analysis was performed. The results of the analyses provide estimated accident frequencies and the potential health effects to workers and members of the public resulting from accidental exposure to radionuclides or nonradioactive toxic materials.

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CONTENTS

TABLES

ACCIDENT ASSESSMENTS FOR IDAHO NATIONAL ENGINEERING LABORATORY FACILITIES

1. INTRODUCTION

The *Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Environmental Impact Statement* (DOE 1995), hereafter referred to as the EIS, evaluates four proposed alternatives for conducting activities at the Idaho National Engineering Laboratory (INEL):

- **Alternative A: No Action**
  Complete all near-term actions identified and continue operating most existing facilities. Serves as benchmark for comparing potential effects from the other three alternatives.

- **Alternative B: Ten-Year Plan**
  Complete identified projects and initiate new projects to enhance cleanup, manage INEL waste streams and spent nuclear fuel, prepare waste fuel for ultimate disposal, and develop technologies for fuel disposition.

- **Alternative C: Minimum Treatment, Storage, and Disposal**
  Minimize treatment, storage, and disposal activities at the INEL to the extent possible (including receipt of spent nuclear fuel). Conduct minimum cleanup and decontamination and decommissioning prescribed by regulation. Transfer spent nuclear fuel and waste from environmental restoration activities to another site.

- **Alternative D: Maximum Treatment, Storage, and Disposal**
  Maximize treatment, storage, and disposal functions at the INEL to accommodate waste and spent nuclear fuel from the U.S. Department of Energy (DOE) complex. Conduct maximum cleanup and decontamination and decommissioning.

The potential impacts of facility accidents under the various alternatives are addressed in Appendix B of Volume 1 and Section 5.14 of Volume 2 of the EIS (DOE 1995). This report provides a compilation of INEL facility accident methodology and results in support of the EIS.

A potential exists for accidents at facilities associated with the treatment, storage, and disposal of radioactive and hazardous materials. Accidents can be categorized into events that are abnormal (for example, minor spills), events a facility was designed to withstand, and events a facility is not designed to withstand (but whose consequences it may nevertheless mitigate). These categories are

- An accident is *bounding* if no reasonably foreseeable accident with greater consequences can be identified within the accident frequency range considered.

- A *preventive feature* is any structure, system, or component that prevents a release of hazardous material from occurring in an accident scenario. Preventive features may include passive barriers such as piping, material containers, material cladding, gloveboxes, or facility structures. It also includes systems or components such as pressure relief valves, monitoring systems for material concentrations with automatic actions to stop or isolate the process, or dilution systems to control explosive or flammable mixtures. *Preventive measures*, as used in this document, include both preventive features and those personnel actions and administrative programs that serve comparable preventive functions.

- A *mitigative feature* is any structure, system, or component that could reduce the consequences from a release of hazardous materials in an accident scenario. Mitigative features may include passive barriers such as dikes, confinement systems, or containment systems; or active systems or components such as air cleanup systems, sump systems, dilution systems to reduce concentration levels, and liquid cleanup system. *Mitigative measures*, as used in this document, includes both mitigative features and those personnel actions and administrative programs that serve comparable mitigative functions.

- An analysis is *conservative* if the net result of its assumptions exaggerates the environmental consequences of the accident being analyzed.

- A *facility worker* is a hypothetical worker located 100 meters (328 feet) from the accident (point of release of radioactive or hazardous materials).

- A *source term* is the amount of radioactive or toxic material released into the environment during a specific scenario. It is not necessarily the total quantity of material present, but is only that material the scenario postulates to be released from the facility.

This document describes the selection of locations or operations for analysis, the development of the bounding scenarios and resulting assumptions about source terms, the selection of computer codes and modeling used to estimate environmental consequences, and the results, including the predicted health effects. The bounding scenarios were developed for specific hazards, such as radioactive material or hazardous materials for operations in all major INEL facility areas. The selected accident scenarios define a bounding envelope of accidents; that is, any other reasonably foreseeable accident at the INEL would be expected to have smaller consequences within a given

2. METHODOLOGY

Methodologies for assessing facility accidents that potentially release radioactive material (Section 2.1) and hazardous material (Section 2.2.) are described in this chapter. These methodology discussions follow a brief discussion of the uncertainties in the environmental consequence analyses.

The calculations have generally been performed in such a way that the estimates of risk provided are unlikely to be exceeded in the event of an accident. The analyses of hypothetical accidents provide more opportunities for uncertainty than those for normal operations, primarily because the calculations must be based on sequences of events and models of effects that have not occurred. In this document, the goal in selecting the hypothetical accidents analyzed has been to evaluate events that would produce effects that would be as severe or more severe than any other accidents that might reasonably be foreseen. The models have attempted to provide estimates of the probabilities, source terms, pathways for dispersion and exposure, and the effects on human health and the environment that are as realistic as possible. However, in many cases, the very low probability of the accidents postulated has required the use of models or values for input that produce estimates of consequences and risks that are higher than would actually occur because of the desire to provide results that will not be exceeded.

As a result of the methods used, the choices of accident scenarios, and the data used in calculations, it is estimated that the net effect is that the calculated risks presented for the accidents in this document are on the order of 10 to 100 times greater than the risks that would be most likely to actually occur.

The use of conservative analyses is not an important problem or disadvantage in this document and in the EIS because all of the alternatives have been evaluated using the same methods and data, allowing a fair comparison of all of the alternatives on this same basis. It should be observed that, even using these conservative analytical methods, the risks for all of the alternatives are small.

2.1. Accidents With Potential Release of Radioactive Material

Radioactive materials are involved in a wide variety of operations at the Idaho National Engineering Laboratory (INEL), including scientific research and engineering development for both domestic and national defense purposes. In the past four decades, the INEL has been the world's most notable research and development center for testing of nuclear power reactor concepts, their fuels, their stability, and their behavior in accidents, as well as a center for the reprocessing of spent nuclear fuel. Radioactive materials encompass potentially valuable resources, such as spent nuclear fuels and various isotopes, but also include waste products ranging in form from contaminated laboratory equipment and metal filings to contaminated trash and liquids. These resources and wastes

- *Category 3.* The hazard analysis shows the potential for only significant localized consequences.

These categories (or the classifications performed under the previous DOE order) were used as a screening threshold. Category 3 (low) hazard facilities were excluded since accidents in these facilities would be bounded by those in Category 2 (moderate) or Category 1 (high) hazard facilities. Those facilities with a hazard classification of 2 or greater were evaluated further. They were ranked on the basis of their total quantities of radioisotopes, their potential for an accident occurring, and their relationship with surrounding facilities. Changes in projected inventories by alternative at the various facilities were considered.

This information combined with reviews of existing safety analysis documentation and discussions with plant personnel confirmed that accidents in the resulting facilities would have the potential of producing bounding consequences.

The process was completed after INEL and DOE personnel further reviewed the selected facilities to ensure all significant radioactive materials in the nonreactor nuclear facilities were considered. A review step allowed reconsideration of facilities previously excluded. In the present analysis, one facility, the INEL Research Center in Idaho Falls, was reconsidered because of the proximity of these laboratories to the public. Refer to Appendix A for additional discussion of accident screening.

2.1.1.2 Determination of Qualitative Likelihood of "Reasonably Foreseeable" Accidents. The purpose of determining the qualitative likelihoods of the "reasonably foreseeable" accidents postulated in this document is to assist in assessing their significance and to provide information to put accidents in perspective relative to each other. The qualitative likelihood was determined by estimating the likelihood of the postulated accidents. Descriptions of the accidents and data obtained from a variety of sources were used to estimate accident likelihood. Once the likelihood was estimated for each accident, they were classified by a likelihood range.

Many schemes for estimating likelihood ranges are available from U.S. Department of Defense, DOE, and commercial applications. The three likelihood ranges chosen, based on the frequency of an accident per facility year, are as follows:

Category	Likelihood range (accidents per year)
Abnormal events	frequency $\geq 1 \times 10^{-3}$
Design basis events	$1 \times 10^{-3} >$ frequency $\geq 1 \times 10^{-6}$
Beyond design basis events	$1 \times 10^{-6} >$ frequency $\geq 1 \times 10^{-7}$

With one exception the mechanism for radioactive materials reaching human receptors as a result of the accidents postulated in this document is by airborne release. (The one exception, a groundwater release, is described in Section 4.1.2.4.) The four commonly accepted exposure pathways (Figure 2.1-1) are as follows:

- External direct exposure from immersion in the passing airborne plume
- External direct exposure from radioactivity deposited on the ground
- Internal exposure from inhalation of respirable fractions of aerosols and suspended particles
- Internal exposure from ingestion of terrestrial food and animal products at the site boundary and beyond.

The Radiological Safety Analysis Computer Program (RSAC-5) (Wenzel 1993) was the computer code chosen for estimating radiation doses resulting from the airborne release of radionuclides. Two other computer codes, ORIGEN2.1 (Croff 1983, RSIC 1991) and Microshield 3.13 (Grove 1988) were used for some accident scenarios to calculate radionuclide inventories as input to RSAC-5. Figure 2.1-2 is a flow sheet showing the interrelationship among the three codes as well as among the various input parameters discussed in this section.

2.1.2.1 RSAC-5 Code. The computer code RSAC-5 was developed, for the DOE-ID Operations Office, by Westinghouse Idaho Nuclear Co., Inc. (WINCO) (Wenzel 1993) and is in the public domain.

RSAC-5 simulates potential radiation doses to maximally exposed individuals or population groups from accidental releases of radionuclides to the environment. From a specified or RSAC-calculated source term, users can calculate the environmental transfer, uptake, and human exposure. Individual doses are determined at specific distances onsite, at the site boundaries, and away from the site for the four exposure pathways. (The ingestion pathway applies only where food is raised locally and potentially consumed there.) Population doses are the product of individual dose and the number of people in the affected population.

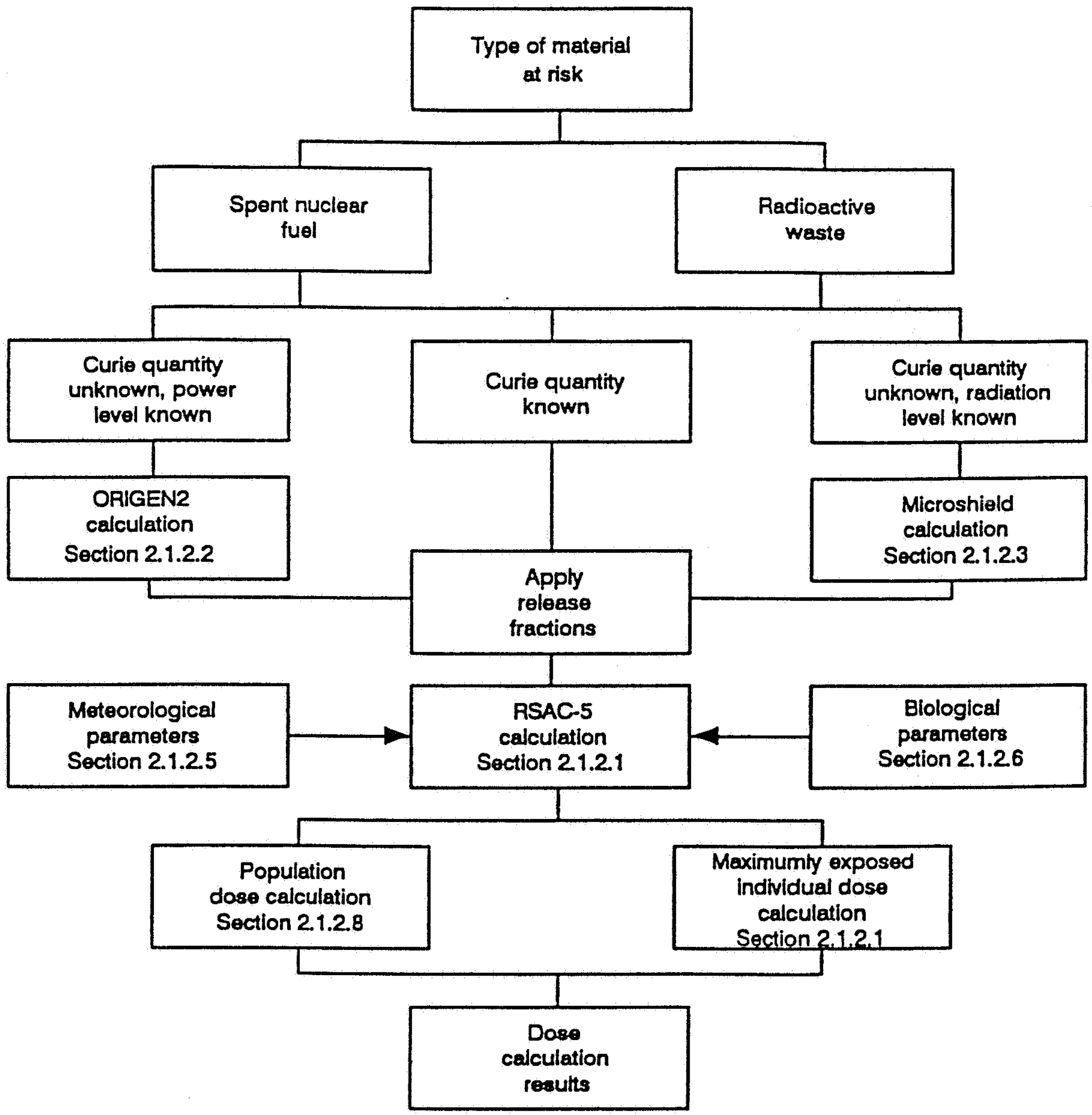

Figure 2.1-2. Flow sheet of codes and input parameters for radiological modeling.

The user has the option of directly entering χ/Q or having the χ/Qs calculated by the code. If code-calculated χ/Qs are chosen, the user must specify the release height, mixing layer height, wind speed, and atmospheric stability class. The code also provides options to calculate plume rise or stack releases. Other code options for calculating atmospheric transport include plume depletion by wet or dry deposition and building wake effects.

2.1.2.1.3 Dose Calculations—As recommended by the International Commission on Radiological Protection (ICRP 1977, 1979), RSAC-5 uses weighting factors for various body organs to calculate a "committed effective dose equivalent" (CEDE) from radioactivity deposited inside the body by inhalation or ingestion. Committed dose[a] equivalents (CDEs) are calculated for the lungs, stomach, small intestine, upper large intestine, lower large intestine, bone surface, red bone marrow, testes, ovaries, muscle, thyroid, bladder, kidneys, and liver. The CEDE is the summation of the CDEs to specific organs weighted by the relative risk to that organ compared to an equivalent whole-body dose.

RSAC-5 calculates an effective dose[b] equivalent (EDE) for the external exposure pathways (immersion in plume, from ground surface contamination) and a 50-year CEDE for the internal exposure pathways (inhalation, ingestion). The sum of the EDE from external pathways and the CEDE from internal pathways is called the "total effective dose equivalent" (TEDE) in this document. The TEDE summation is performed external to RSAC-5.

Dose calculations may be performed using either code default parameters or user-supplied parameters for biological uptake (see Section 2.1.2.6). Internal CDEs are calculated using dose conversion factors from DOE/EH-0071 (DOE 1988a). Ingestion uptake is calculated using models described in NRC Regulatory Guide 1.109 (NRC 1977a). Standard ingestion constants are provided in the code; however, one may alter any of the constants to fit site-specific conditions. The external EDE for the ground surface and air immersion (semi-infinite plume model) pathway is calculated using dose conversion factors from DOE/EH-0070 (DOE 1988b).

Doses may be calculated for an individual at a specified receptor location out to 100 kilometers (62 miles) or for a population within a 80-kilometer (50-mile) radius of the point of release. Population doses are determined by calculating an average individual TEDE at 16-kilometer (10-mile) radial intervals of a compass sector and then multiplying by the number of people to whom that average TEDE applies. (See Figure 2.1-3 in Section 2.1.2.8.) The 16 compass sectors radiate in

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a. Everyday examples of committed doses occur when medical technicians administer radionuclides into patients for treatment or diagnostic purposes. These radionuclides are "committed" to the body and continue to irradiate tissues until they are expelled from the body or decay to insignificant concentrations.

b. Everyday examples of effective doses occur when medical technicians use x-rays for diagnostic purposes. Radiation from this external exposure pathway passes through the body and irradiates tissues only during the passage.

2.1.2.2 ORIGEN2.1: Isotope Generation and Depletion Code. ORIGEN (Croff 1983, RSIC 1991) is a computer code system for calculating the buildup, decay, and processing of radioactive materials (fission products, actinides and activation products). It is one of two computer codes recommended by the NRC (NRC 1977b) for calculating the radioactivity initially present and later produced in an inadvertent nuclear chain reaction in a fuel reprocessing plant.

ORIGEN2 (Croff 1983) is a revised version of ORIGEN and incorporates updates of the reactor models, cross sections, fission product yields, decay data, and decay photon data, as well as the source code. ORIGEN2 is thus a reactor physics code that provides various nuclear material characteristics in a variety of engineering units, while employing a relatively unsophisticated neutronics calculation. The ORIGEN2 databases contain 130 actinides, 850 fission products, and 720 activation products (a total of 1700 nuclides, including some that appear in more than one category).

ORIGEN2.1 (RSIC 1991) replaces ORIGEN2 and includes additional libraries for standard and extended burnup calculations for pressurized water reactors and boiling water reactors. The INEL version of ORIGEN2.1 is run on the INEL Cray X-MP computer. It includes a special radionuclide library for INEL's Advanced Test Reactor.

For this document, ORIGEN2.1 was used in accident analyses involving significant contribution of actinides and activation products to the radioactive source term associated with spent fuel and inadvertent nuclear chain reaction accidents (for example, see Section 3.1.2.1). The radioactivity of each such radionuclide (in curies) in the material damaged by the accident, as calculated by ORIGEN2.1, was multiplied by the appropriate release fraction and supplied as input to subsequent RSAC-5 calculations. (RSAC-5 calculates only the contribution of fission products to the radioactive source term.) Secondarily, the ORIGEN2.1 results regarding the radioactivity of the fission product source term were used as a check of the comparable RSAC-5 results.

2.1.2.3 Microshield 3.13. Microshield (Grove 1988) is a radiation shielding code developed for analysis of shielding design, container design, and selection of temporary shielding. Another use of Microshield, employed in some of the accident analyses in this document, is the calculation of source strength on the basis of radiation measurements from a shielded source of known material and dimensions. This calculation is an iterative process of estimating values of the source strength until the measured radiation values are matched by the calculation.

This program is a microcomputer adaptation of the main frame code ISOSHLD (Engle 1966), a public domain code originally written in the early 1960s. Microshield retains only the solution algorithms and physical data from ISOSHLD. User interaction with it conforms to American National Standards Institute (ANSI) 6.6.1 (ANSI 1979).

2.1.2.4.1 Damage Ratio—The damage ratio is the fraction of material exposed to the effects of the energy/force/stress generated by the postulated event. For the bounding accident scenarios discussed in this document, the damage ratio is one unless otherwise specified.

2.1.2.4.2 Airborne Release Fraction—The airborne release fraction is the fraction of the material that is made airborne due to the accident. Values from generic DOE guidance (Elder et al. 1986, DOE 1993c) are used in this document unless more specific information is provided in source documents applicable to a particular accident scenario. These values are summarized in Table 2.1-19 in Section 2.1.4

2.1.2.4.3 Respirable Fraction—The respirable fraction is the fraction of the material, with particle sizes less than 10 microns (DOE 1993c) that could be retained in the respiratory system following inhalation. It is applied only to the source term for the inhalation pathway.

2.1.2.4.4 Leak Path Factor—The leak path factor accounts for the action of removal mechanisms, such as containment systems, filtration, deposition, etc., to reduce the amount of airborne radioactivity that is ultimately released to occupied spaces of the facility or to the environment. An leak path factor of one is assigned for a major failure of confinement barriers.

2.1.2.5 Meteorological/Dispersion Parameters. For accidents initiated within the INEL, radiological doses are calculated not only for the general population, but also usually at three locations: (a) for facility workers within the originating facility area (for example, Idaho Chemical Processing Plant), at 100 meters (328 feet) from source, (b) at the nearest public access to the accident location, and (c) at the nearest INEL site boundary. The nature of the release and the meteorological dispersion parameters associated with these three locations are discussed below. Applicable parameters are summarized in Table 2.1-20 in Section 2.1.4. A qualitative assessment of representative accidents for workers less than 100 meters from the source is given in Appendix C.

The release duration is specific to each accident scenario. The rate of release is assumed to be linear (constant with time) unless otherwise specified. With a linear release rate, the release coefficient is the inverse of the release duration.

The RSAC-5 code has three sets of dispersion coefficients built in for calculation of atmospheric dispersion. Hilsmeier-Gifford coefficients are used for desert terrains (such as the INEL) for effluent releases from a few minutes to 15 minutes duration (Clawson et al. 1989). Markee coefficients are used for desert terrains for effluent releases from 15 to 60 minutes duration (Clawson et al. 1989). Pasquill-Gifford coefficients were developed from the Prairie Grass experiments for effluent releases from 10 to 60 minutes duration (Slade 1968, NRC 1982). The Hilsmeier-Gifford and Markee coefficients were derived from experiments at the INEL and are best suited for modeling observed dispersion factors would be larger (less conservative) than the 95 percent value, and 95 percent of the values would be less than or equal to it. Parameters for both 50 percent (average conditions) and 95 percent meteorological conditions are summarized in Table 2.1-20 in Section 2.1.4.

Workers within the facility area and individuals at the nearest public access and nearest site boundary are assumed directly downwind from the accident location. For population doses the wind direction is constrained to the directions with the highest consequences for the general population, as discussed in Section 2.1.2.8.

2.1.2.6 Biological Parameters.

2.1.2.6.1 Inhalation Pathway Parameters—Inhalation parameters are summarized in Table 2.1-1 and discussed below. These parameters are the same for all the radiological scenario analyses in this document.

Table 2.1-1. Inhalation parameters used in dosimetry calculations.

Inhalation parameter	Facility worker	Nearest public access	Nearest site boundary
Breathing rate (m^3/s)	3.33×10^{-4}	2.66×10^{-4}	2.66×10^{-4}
Alternate clearance class for inhalation[a]	Pu - week Sr - year	Pu - week Sr - year	Pu - Week Sr - Year

a. All other clearance classes are RSAC-5 default values.

Standard compartmental models for transport of nuclides are assumed for transport of inhaled or ingested nuclides, and standard physical parameters such as body weight are used in all calculations (ICRP 1979). Breathing rates are assumed to be 3.33×10^{-4} cubic meters per second for exposures at controlled areas like the ICPP facility area [DOE Order 5480.11 (DOE 1991b)] and 2.66×10^{-4} cubic meters per second for uncontrolled area like public highways inside the INEL site and at the nearest site boundary [DOE Order 5400.5 (DOE 1990a)].

Following inhalation, the transfer rate of an isotope from the pulmonary section of the lungs is dependent upon the isotope's chemical form. The conceptual lung model defined in ICRP Publication 30 (ICRP 1979) accounts for this dependency by grouping chemical forms for each isotope into broad categories, or clearance classes. These clearance classes of D, W, or Y indicate that pulmonary clearance half-times are on the order of days, weeks, or years, respectively.

- After the accident is over and the airborne release is terminated, workers are evacuated to buses in a nearby parking lot. During transit from buildings to the buses, workers are exposed to radioactivity deposited on the ground surface for a limited time (a maximum of 15 minutes).

- Workers are exposed to radioactivity via the inhalation, air immersion, and ground surface pathways only. Ingestion of food plants or animals grown onsite at INEL is not expected for facility workers.

The following assumptions apply to the maximally exposed individual at the nearest public access:

- The nearest public access to the location of an accident is usually a public highway (e.g., for ICPP, U.S. Highway 20/26 near the EBR-I National Historic Monument is approximately 5.9 kilometers (3.7 miles) from the ICPP facility area).[a] This location is within the INEL site boundaries and is patrolled by the INEL Security force. In the event of an accident with potential impacts outside the complex boundary, public access to the highway would be controlled by roadblocks established by INEL Security and State Highway Patrol. It is conservatively assumed that a motorist could be on such a highway for up to two hours before being evacuated by INEL Security personnel.

- A member of the public on such a public highway directly downwind of an accident location would be exposed to radioactivity via the inhalation, air immersion, and ground surface pathways only. Consumption of food plants or animals grown onsite is not expected for a member of the public temporarily on INEL. For the inhalation and air immersion pathways, exposure time to the plume would be for the entire release duration up to a maximum of two hours. Exposure time to radioactivity deposited on the ground surface would be a maximum of two hours.

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a. For nearest public access and nearest site boundary receptor distances from selected INEL sources, see Table 2.1-2 (Leonard 1992).

The following assumptions apply to the maximally exposed individual at the nearest site boundary:

- A hypothetical member of the public resides at the INEL nearest site boundary (e.g., for ICPP, approximately 14 kilometers or 22.5 miles). This individual grows crops and raises animals for personal food consumption. It is assumed that the wind blows directly toward this person and this person's land when the accident occurs, and that this person receives no warning of the accident.

- This hypothetical member of the public at the nearest site boundary directly downwind of the accident would be exposed to radioactivity via the inhalation, air immersion, ingestion, and ground surface pathways. For the inhalation and air immersion pathways, exposure time to the plume would be for the entire release duration. Crops and grazing land are exposed for the entire duration of plume passage.

- Food contaminated by the accidental release of radioactivity is assumed to be ten percent of the hypothetical individual's diet during the ensuing year. This percentage is considered consistent with normal practices that would reduce contamination, such as sprinkler irrigation and washing of vegetables. It does not take credit for interdictive measures, such as enforced limits on consumption unless exposures reach values where protective action guidelines are exceeded (see Section 2.1.3.4).

- Exposure time to radioactivity deposited on the ground surface would be a maximum of 70 percent of each year because the individual could reasonably be expected to spend, on the average, at least 30 percent of each day indoors and shielded from ground surface radioactivity.

2.1.2.8 Population Dose Estimates. The RSAC-5 option for calculating population doses (in person-rem) involves determining a TEDE (total effective dose equivalent, in rem) for an average individual at several locations within a 80-kilometer (50-mile) radius and multiplying that TEDE by the number of persons to whom it applies. The TEDE calculation is similar to that for the maximum exposed individual at the nearest site boundary, with some limitations and exceptions:

- For the population option, RSAC-5 limits the radionuclide inventory to 100 entries. For scenarios with more than 100 nuclides, such as those for inadvertent nuclear chain reactions, a screening step is performed. Only those nuclides that produce an EDE or CEDE greater than one millirem for any one of the four pathways at any of the three locations are included.

- In the ingestion pathway, the consumption rates are reduced (Rupp 1980) as mentioned in Section 2.1.2.6.2.

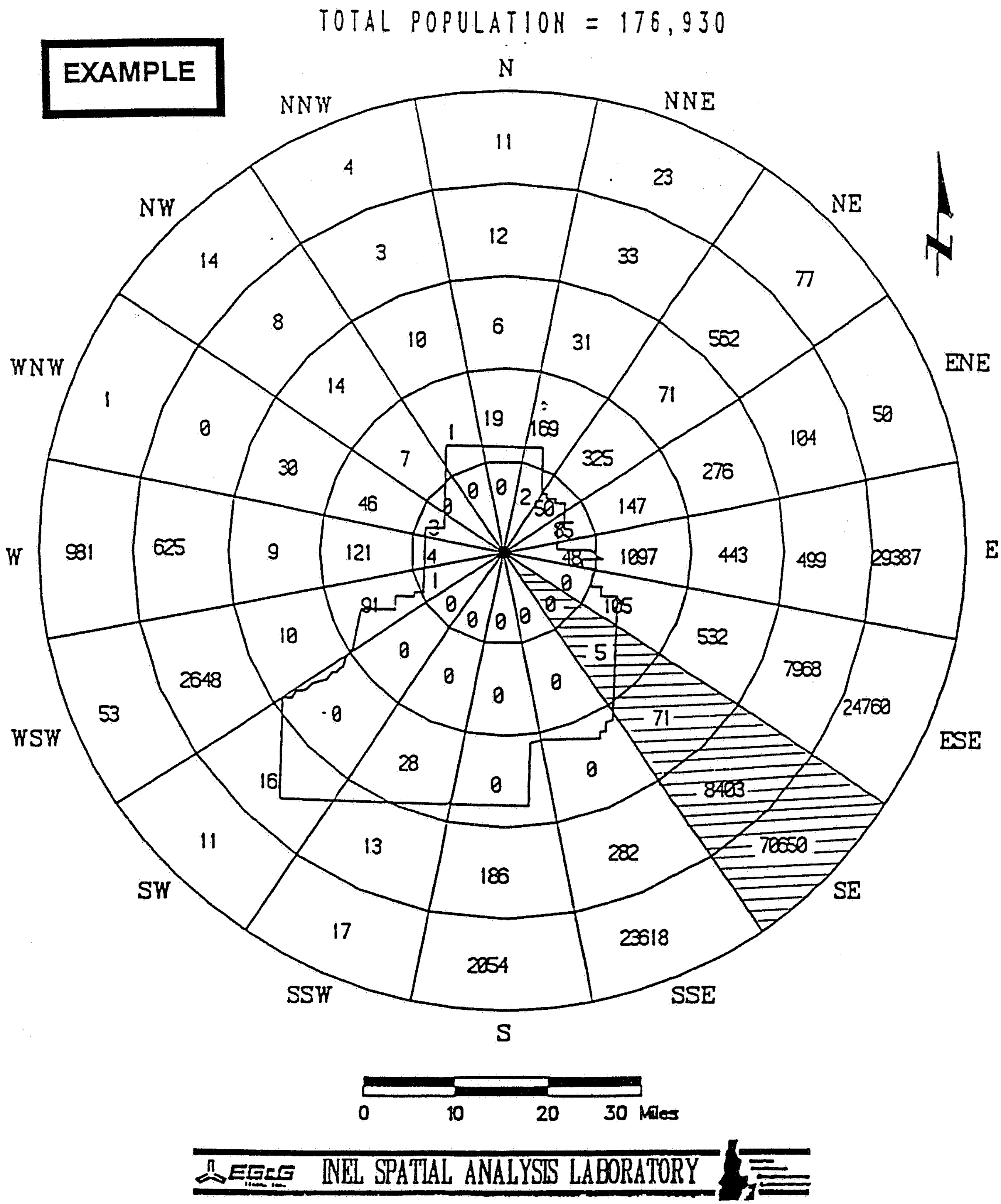

Figure 2.1-3. Example of population input to population dose calculation (projection based on 1990 census data).

2.1.3 Perspectives on the Bounding Accident Assessments

Because the set of bounding accidents considered in this document are neither likely to occur, nor to occur with such large consequences, this section provides perspective about the accident assessments from the following aspects:

- A review of previously considered assessments from safety analysis reports for selected INEL facilities (ranging from operational upsets to design basis accidents)
- A review of accidents that have occurred at INEL
- A discussion of preventive features at INEL facilities intended to reduce the likelihood of accidents
- A discussion of the mitigative measures at INEL facilities intended to reduce the consequences of accidents.

2.1.3.1 Previously Considered Accidents. Safety analysis reports have been prepared for INEL environmental remediation and waste management facilities and operations where significant potential hazards are present. Tables 2.1-4 through 2.1-16 summarize the accidents analyzed in the safety analysis report for each major INEL environmental remediation and waste management facility, the accident frequency, and the analyzed consequences of each accident scenario. The preventive and mitigative measures assumed in the safety analysis report are also summarized.

2.1.3.2 INEL Accident History. This section provides a brief discussion of historical accidents at the INEL environmental restoration and waste management facilities. Information on accidents that have occurred at INEL facilities is based on review of safety analysis reports, and the INEL Historical Dose Evaluation Project (DOE-ID 1991b). The airborne pathway is the principal pathway by which radioactive materials released on the INEL can reach an offsite member of the public. Accidents resulting in the most significant airborne releases of radioactivity are summarized in Table 2.1-17. Although accidental releases to the soil have occurred on the INEL as discussed below, there has been no significant dose to an offsite member of the public through the Snake River aquifer to date (DOE-ID 1991b).

The accident with the highest onsite and offsite consequences in the operating history of the INEL was the SL-1 reactor accident in January 1961. An inadvertent nuclear chain reaction and steam explosion at the reactor resulted in the death of three workers at the plant site and release of 1,100 curies of radioactivity. The dose to a maximally exposed member of the public at the INEL boundary has been estimated at 3 millirem as a result of the release (DOE-ID 1991b).

Table 2.1-6. Summary of accidents analyzed for high-level liquid waste storage tanks at Idaho Chemical Processing Plant.[a]

Analyzed accident	Frequency	Consequences[b]
Tank failure	Not specified	Soil contamination, possible groundwater contamination
Transfer line failure	Not specified	Soil contamination, possible groundwater contamination

a. Source: ICPP Plant Safety Document, Section 4.2 (WINCO 1990).

b. No mitigative measures assumed in the analyses.

Table 2.1-7. Summary of accidents analyzed for Atmospheric Protection System at Idaho Chemical Processing Plant.[a]

Analyzed accident	Frequency	Consequences[b]
Stack toppling	Not specified	Some release possible, health effects not quantified
Filter fire (maximum postulated accident)	Not specified	1 rem worker, 70 mrem at nearest site boundary
Explosion	Not specified	Some release possible, health effects not quantified
In-plant fire	Not specified	Release unlikely, health effects not quantified
Ventilation duct rupture	Not specified	Some release possible, health effects not quantified
Loss of electricity	Not specified	Release unlikely, health effects not quantified

a. Source: ICPP Plant Safety Document, Section 4.3 (ENICO 1980b).

b. Mitigative measures assumed in the analyses: Release through stack at the 76-meter (250-foot) elevation.

Table 2.1-10. Summary of accidents analyzed for Test Area North.[a]

Analyzed accident	Frequency	Consequences[b]
Test Area North Hot Shop		
Fuel cladding failure	<1E-06/yr	63 rem to the thyroid at nearest site boundary
Specific Manufacturing Capability		
Nitric acid spill	Not specified	Exceeds ERPG-2 within 11 meters
Depleted uranium building fire	<1E-06/yr	2 rem at Idaho Highway 33

a. EG&G Idaho (1991), Sanford (1993), Janke (1988), Janke and Girton (1987).

b. No mitigative measures assumed in the analyses.

Table 2.1-11. Summary of accidents analyzed for Central Facilities Area.[a]

Analyzed accident	Frequency	Consequences[b]
Hazardous Waste Storage Facility		
Vehicle accident	<1E-06/yr	Not quantified
Accident at adjacent recyclable oil collection tank	<1E-06/yr	Not quantified
Small aircraft/helicopter crash	<1E-06/yr	Not quantified
Explosion/fire in stored waste (maximum hypothetical accident)	5E-05/yr	Not quantified, possible phosgene release.
Engineering Research and Applications Laboratory (CFA-625)		
Full building fire	1E-04 to 1E-06/yr	Chemical release, below time-weighted average (TWA) at 100 m
CFA Sewage Treatment Plant		
Chlorine gas release	Not specified	Possible health effects within 100 m
Radiation and Environmental Sciences Laboratory (RESL)		
Chemical release	Not specified	Possible health effects within 100 m
Decontamination and Decommissioning Activities		
Fire, structural failure, others	Not specified	Impacts to worker, <2 mrem maximum offsite individual (nearest site boundary)

a. Sources: EG&G Idaho (1990b), DOE (1992b), EG&G Idaho (1992d).

b. No mitigative measures assumed in the analyses.

Table 2.1-13. Summary of accidents analyzed for Mixed Waste Storage Facility at the Waste Reduction Operations Complex.[a]

Analyzed accident	Frequency	Consequences[b]
Minor radiological/toxic waste incident	1E+00 to 1E-02/yr	Release possible, health effects not quantified
Minor industrial accident	1E+00 to 1E-02/yr	Minimal personnel injury, $50,000 equipment/facility damage
Personnel exposure	1E-02 to 1E-04/yr	Worker exposure within DOE 5480.1B limits (DOE 1986)
Major fire	1E-02 to 1E-04/yr	Release possible, health effects not quantified
Major industrial accident	1E-02 to 1E-04/yr	Personnel injury or death, $250,000 equipment/facility damage
Loss of commercial power while handling mixed waste	1E-02 to 1E-04/yr	Release possible, health effects not quantified
Compressed gas accident	1E-02 to 1E-04/yr	Release unlikely, health effects not quantified
Biological contamination	1E-02 to 1E-04/yr	Release unlikely, health effects not quantified
Tornado	1E-04 to 1E-06/yr	Release possible, health effects not quantified
Severe earthquake	1E-04 to 1E-06/yr	Release possible, health effects not quantified
Potentially fatal radiation exposure through internal exposure/direct radiation	1E-04 to 1E-06/yr	Worker death
Toxic exposure (maximum hypothetical event)	Not specified	Possible phosgene exposure, health effects not quantified
Missile generation (maximum hypothetical event)	Not specified	Equipment/facility damage
Fire in flammables storage area (maximum hypothetical accident)	Not specified	100 m: 15 mrem to bone Nearest site boundary: <1 mrem

a. Source: EG&G Idaho (1990a).

b. No mitigative measures assumed in the analyses.

Table 2.1-16. Summary of accidents analyzed for Hot Fuel Examination Facility at Argonne National Laboratory-West.[a]

Analyzed accident	Frequency	Consequences[b]
Inadvertent nuclear chain reaction	<1E-06/yr	Not calculated
Sodium fire	<1E-06/yr	Not calculated
Cell breach	<1E-06/yr	0.668 rem at nearest site boundary

a. Source: Adams et al. (1975).

b. Mitigative measures assumed in the analyses: Evacuation of a member of the public at the nearest site boundary after two hours.

Table 2.1-17. Significant accidental radioactivity releases to the atmosphere at the INEL.[a]

Date	Description	Area (present name)	Total release (Ci)	Effective radiation dose (mrem)[b]	Onsite fatalities
10/16/59	Nuclear chain reaction	ICPP[c]	37,000	1.1	0
1/3/61	SL-1 reactor accident	ARA[d]	1,100	3.0	3[e]
10/17/78	Nuclear chain reaction	ICPP	620	< 0.1	0
1/25/61	Nuclear chain reaction	ICPP	120	< 0.1	0
10/30/58	FECF[f] filter break	ICPP	39	0.11	0

a. Source: DOE-ID (1991b).

b. Maximum dose to a member of the public.

c. ICPP - Idaho Chemical Processing Plant.

d. ARA - Auxiliary Reactor Area.

e. The accident occurred when a safety rod was manually lifted beyond the safe limit, resulting in an inadvertent nuclear chain reaction and steam explosion. Three workers were present at the facility at the time of the accident; two were killed in the explosion and the third was fatally injured.

f. FECF - Fuel Element Cutting Facility.

Accidental releases have occurred from the ICPP Atmospheric Protection System or other ventilation systems at ICPP. The most significant release occurred in 1958 when filters in the CPP-601 Fuel Element Cutting Facility (FECF) failed during decontamination operations. Approximately 100 curies of particulate activity was released over an area of approximately 81 hectares (200 acres) in the vicinity of ICPP. An estimated 39 curies became airborne, resulting in an estimated dose to a maximally exposed offsite individual of 0.11 millirem (DOE-ID 1991b).

Unplanned releases of ruthenium-106 have occurred from the ICPP main stack. Releases have been estimated at 1 curie or less (DOE-ID 1991b).

Three fires have occurred at the Radioactive Waste Management Complex (RWMC). Two occurred in 1966 in exposed waste material in trenches, thought to be caused by alkali metals in disposed waste. The third fire occurred 1970 in a drum of stored waste from the Rocky Flats Plant, postulated to have been caused by radiant solar heating of the black drum surface. Monitoring and accident recovery activities from the fires indicated that releases and spread of radionuclides was undetectable (EG&G Idaho 1986).

One accident involving a spill and release of radioactive material occurred in 1978. In a handling accident, a drum was penetrated by a forklift tine, spilling a portion of the drum contents. The spilled waste was immediately contained, and no detectable airborne release of radionuclides occurred (EG&G Idaho 1986).

The RWMC Subsurface Disposal Area has been flooded by local runoff in 1962, 1969, and 1982. During each of these events, water may have made direct contact with waste, causing potential leaching of contaminants into the soil (EG&G Idaho 1986).

2.1.3.3 Preventive Measures. This analysis of radiological accidents considers accidents that cause releases greater than those expected from routine operations. As discussed below, preventive measures (see definition in Section 1) at two points in the postulated accident scenarios could preclude releases.

First, preventive measures can *prevent some accident initiating events from occurring*. Examples of measures to prevent accident initiating events include the following:

- The safety analysis report process, which identifies potential accidents and ensures that prudent specific preventive measures are implemented

- The practice of using ventilating systems to prevent combustible or explosive mixtures from accumulating in confined spaces and becoming an initiating event

- DOE Emergency Operations Center
- County and State Emergency Command Centers
- Medical, health physics, and industrial hygiene specialists
- Protective clothing and equipment (respirators, breathing air supplies, etc.).

The radiation doses estimated in this document for the various radiological accident scenarios are the doses that would be received by the population if only limited protective actions were taken. INEL has detailed plans for responding to accidents of the type described here, and the response activities would be closely coordinated with state and local officials. INEL personnel are trained and drilled in the protective actions to be taken if a release of radioactive or otherwise toxic material occurs.

In most accidents, the time it takes the gaseous plume to pass the receptor is less than the assumed two-hour evacuation time for a motorist at the nearest public access highway. However, in a few accidents, the assumed release time is 24 hours, so credit is taken for evacuation in reducing the dose at the nearest public access.

For the offsite population, the need for any protective action would be based on the predicted radiation doses. The emergency response would be based on the guidance provided in the protective action guides developed by the U.S. Environmental Protection Agency (EPA 1991), as summarized in Table 2.1-18.

Interdiction activities by INEL accident recovery personnel are expected to take place following the accident to limit doses to offsite individuals at risk. This interdiction can limit ingestion exposure so that the maximally exposed individuals will derive much less than the assumed 10 percent of their diet from locally grown crops and livestock.

Generally, according to the EPA guidance document (EPA 1991), no protective action is required when the projected doses are less than 1 rem to the whole body or less than 5 rem to the thyroid, but radiation levels should be monitored, and an advisory to seek shelter may be issued. For whole-body doses of 1 to less than 5 rem and thyroid doses of 5 to less than 25 rem, the public should be warned to seek shelter, access to the contaminated area should be controlled, and evacuation should be considered unless constraints make it impractical. For whole-body and thyroid doses higher than 5 and 25 rem, respectively, evacuation would be mandatory, access to the contaminated area would be controlled and radiation levels would be monitored. As shown in Table 2.1-18, higher doses are allowed for emergency workers.

The underlying principle for the protective action guides is that under emergency conditions all *reasonable* measures should be taken to minimize the radiation exposure of the general public and emergency workers. In the absence of significant constraints, protective actions may be implemented when projected doses are lower than the ranges given in the protective action guides.

The Food and Drug Administration also recommends protective actions to protect the public health from food contamination resulting from radiation incidents (FDA 1982). *Preventive* protective action guides are projected dose commitments of 1.5 rem to the thyroid, or 0.5 rem to the whole body, bone marrow, or any other organ for which actions should be taken to prevent or reduce the radioactive contamination of human food or animal feeds. *Emergency* protective action guides are projected dose commitments of 15 rem to the thyroid or 5 rem to the whole body, bone marrow, or any other organ. At this level, responsible officials should isolate food containing radioactivity to prevent its introduction into commerce and determine whether condemnation or another disposition is appropriate. Such interdiction activities by INEL personnel are discussed previously in this section.

2.1.4 Description of Radiological Accident Scenarios and Generic Parameters

Following the selection process described in Section 2.1.1, evaluation of potential bounding accident scenarios (including such related elements as meteorological parameters, dispersion parameters, dose estimates, and emergency response and protective actions) led to a selection of ten bounding accident scenarios involving radioactive materials. Each of these radiological scenarios is described in the following sections according to five major headings:

- Description of Accident
- Development of Radioactive Source Term
- Dose Calculations and Results
- Preventive and Mitigative Measures
- References.

The contents of these sections and the generic parameters used follow.

Description of Accident provides a basis for accident selection, documents the facility and postulated accident scenarios considered, discusses possible initiating events, and lists the major events of the accident scenario. A qualitative assessment of scenario likelihood is developed.

Development of Radioactive Source Term describes the assumptions that apply to the development of the resulting source term. Specifically, it discusses the various multipliers (defined in Section 2.1.2.4) that convert the material at risk to the source term.

Dose Calculations and Results relates the computer modeling (see Section 2.1.2) to the specific accident scenario, and documents results therefrom. Specifically, these subsections

- Describe assumptions and unique input parameters (other than the source term) to the computer model

- Document the computer model output in terms of exposure to radionuclides for individuals and for the population within an 80-mile (50-mile) radius

- Assess the potential for health effects.

Unless otherwise specified, the meteorological/dispersion parameters summarized in Table 2.1-20 are used in the dosimetry calculations for specific accident scenarios. Unless otherwise specified, nominal exposure times to receptors are as summarized in Table 2.1-21. Under some circumstances, facility worker exposures could be either greater or lesser than these nominal values.

Preventive and Mitigative Measures discuss those measures that would be expected to make the specific accident scenario more unlikely or have lower consequences than the assessed consequences. A general discussion of such measures is given in Section 2.1.3.

Table 2.1-21. Exposure times used in dosimetry calculations.

Exposure	Facility worker (100 m)	Nearest public access	Nearest site[a] boundary
To plume[b]	5 min.[c]	100% of release time up to 120 min.[c]	100% of release time
To ground surface[d]	20 min.[e]	120 min.[e]	0.7 yr
To food[e]	Not applicable	Not applicable	1 yr[e]

a. For external direct exposure and for internal exposure from inhalation.

b. For external direct exposure.

c. For internal exposure from ingestion of contaminated food grown at nearest site boundary (10 percent of total food consumption).

d. Applies to most accident scenarios; deviations identified in specific accident descriptions.

e. Nearest site boundary values also used in population dose calculations.

As part of the initial screening, facilities were assigned classifications on the basis of the chemical inventories provided in the SARA list of Extremely Hazardous Substances. Final hazard classifications were based on the reportable chemical quantities within the facilities, National Environmental Policy Act (NEPA) classifications of chemicals stored at the facilities, and the potential consequences of mixing chemicals during an accident. Reviews of existing safety analysis documentation and discussions with plant personnel were used to develop reasonably foreseeable accident scenarios. Refer to Appendix A for additional discussion of accident screening.

2.2.1.2 Determination of Qualitative Likelihood of "Reasonably Foreseeable" Accidents. The method of making this determination is the same as that described in Section 2.1.1.2 for radiological accidents.

2.2.2 Methodology to Estimate Toxic Chemical Exposures

Factors such as receptor locations, terrain, meteorological conditions, release conditions, and characteristics of the chemical inventory are required as input parameters for hand calculations or computer codes to determine human exposure from airborne releases of toxic chemicals.[a] This section gives a general narrative about these input parameters with degrees of conservatism noted, and describes the computer models used to perform exposure estimates. Generic input parameters used in the bounding accident analyses are summarized in Table 2.2-1 in Section 2.2.4. Specific supplementary information is provided in the appropriate subsection of each accident scenario described in that section.

EPIcode™ is the computer code chosen for estimating airborne concentrations resulting from most releases of toxic chemicals (Homann 1988).

2.2.2.1 EPIcode™. Like RSAC, EPIcode™ uses the well-established Gaussian Plume Model to calculate the airborne toxic chemical concentrations usually at the same receptor locations as RSAC; i.e., facility worker, nearest public access, nearest site boundary, and nearby communities. The EPIcode™ library contains information on over 600 toxic substances listed in ACGIH (1987). The types of releases that can be modeled and associated input parameters are also discussed below. Applicable release and dispersion parameters are summarized in Table 2.2-1 (Section 2.2.4).

The continuous release models require specifying the source term as an ambient concentration and a release rate. For term releases, the user specifies the release duration and the total quantity of material released.

a. Computer modeling for the only toxic chemical scenario involving potential groundwater contamination is documented under that scenario (Section 7.1.2.5--rupture of high-level waste tank at the Idaho Chemical Processing Plant).

- Adjusts the wind speed for release height

- Depletes the plume as a function of downwind distance

- Adjusts the standard deviations of the crosswind and vertical concentrations for brief releases.

As output, EPIcode™ can generate data plots of mean toxic chemical concentration (during a specified averaging time) as a function of downwind distance. From these graphs and numerical output, the concentrations at 100 meters (328 feet) (the shortest distance for which EPIcode™ calculates), at co-located facilities, at the nearest public access, at the nearest site boundary, and at nearby communities are determined and evaluated for health effects (see Section 2.2.2.2).

EPIcode™ was selected as the computer code for release analysis of chemicals amenable to Gaussian modeling after comparison with a number of codes, primarily CHARM and ARCHIE. It was judged easier to use for this simple application than either the more sophisticated, proprietary CHARM code or the comparable, public domain ARCHIE code. The SLAB code had previously been selected by INEL as the most appropriate of the refined dispersion models (such as CHARM) for modeling special case releases, such as dense gas dispersion, where negative buoyancy effects must be considered. However, because chemical accident scenarios involving dispersion of denser-than-air gases were not considered in this analysis, the SLAB model was not used. EPIcode™ was judged to be a satisfactory code for the inventory of chemicals analyzed.

2.2.2.2 Health Effects. Hazardous constituents dispersed during an accident could induce adverse health effects among exposed individuals. This possible impact is assessed by comparing the airborne concentrations of each substance at specified downwind receptor locations to standard accident exposure guidelines for chemical toxicity.

Where available, Emergency Response Planning Guideline (ERPG) values are used for this comparison. ERPG values are estimates of airborne concentration thresholds above which one can reasonably anticipate observing adverse effects (Rusch 1993). ERPG values are specific for each substance, and are derived for each of three general severity levels:

> *ERPG-1:* The maximum airborne concentration below which it is believed that nearly all individuals could be exposed for up to 1 hour without experiencing other than mild transient adverse health effects or perceiving a clearly defined objectionable odor.
>
> *ERPG-2:* The maximum airborne concentration below which it is believed that nearly all individuals could be exposed for up to 1 hour without experiencing or developing irreversible or other serious health effects or symptoms that could impair their abilities to take protective action.

procedures for required clothing (rubber boots, gloves, face shields, eye protection) that can mitigate the effects of potential releases of toxic materials. Procedures for handling of toxic materials may also require provisions for installing mitigative devices such as eye wash stations and emergency showers. As discussed in Section 2.1.3.4, INEL facilities employ emergency response programs to mitigate impacts of potential toxic chemical accidents to workers and the public. Emergency planning, emergency preparedness, and emergency response programs are in place and involve established resources such as warning communications, fire departments, and emergency command centers.

2.2.4 Description of Toxic Chemical Accident Scenarios and Generic Parameters

Following the selection process described in Section 2.2.1.1, evaluation of potential accident scenarios (including such relative elements as meteorological parameters, dispersion parameters, exposure estimates, and emergency response and protective actions) led to a selection of several bounding accident scenarios involving toxic materials. Those scenarios are described in Section 7 according to five major headings, similar to those described in Section 2.1.4 for the scenarios involving radioactive materials:

- Description of Accident
- Development of Toxic Chemical Source Term
- Exposure Calculations and Results
- Preventive and Mitigative Measures
- References.

The following paragraphs discuss the material provided under these major headings.

Description of Accident provides a basis for accident selection, documents the facility and postulated accident scenarios considered, discusses possible initiating events, and lists the major events of the accident scenario. A qualitative assessment of scenario likelihood is developed.

Development of Toxic Chemical Source Term describes the assumptions that apply to the development of the resulting source term. Specifically, it discusses the various multipliers that convert the material at risk to the source term. The concepts and assumptions associated with these multipliers are the same as for radioactive materials (see Sections 2.1.2.4 and 2.1.4).

Exposure Calculations and Results relates the computer modeling (see Section 2.2.2) to the specific accident scenario and documents the results of the modeling as follows:

- Presents the computer model used

- Defines the source term input as developed in the previous section

Table 2.2-1. Release and dispersion parameters used for calculating hazardous chemical concentrations resulting from accident scenarios at the Idaho National Engineering Laboratory.[a]

Meteorological/dispersion parameter	Facility worker	Co-located facilities and nearest public access	Nearest site boundary
Receptor distance (m)	100	Specific[b]	Specific[b]
Wind velocity (m/s)	0.5[c,d]	0.5/2.0[c,d,e]	2.0[c,d]
Release elevation[c](m)	0	0	0
Wind stability class[c,d]	F	F	F
Deposition velocity[f] (m/s)			
Solids	0.01	0.01	0.01
Gases/vapors/liquids	0.001	0.001	0.001
Unspecified	0.001	0.001	0.001
Release duration[b]	Specific	Specific	Specific
Release area[g]	Point	Point	Point

a. To convert from meters to feet, multiply by 3.28.

b. Specific to accident scenario.

c. Applies to most accident scenarios; deviations identified in specific accident descriptions

d. Worst-case meteorological conditions are calculated for some scenarios by optional routine.

e. 0.5 meters per second for less than or equal to 2 kilometers from source; 2.0 meters per second for greater than 2 kilometers.

f. Applies to materials (element and physical state) included in specific source terms.

g. Unless area-release calculational option is used.

3. SPENT NUCLEAR FUEL ACCIDENTS

Section 3.1 estimates the consequences for the selected spent nuclear fuel accidents assuming Alternative A is chosen. Section 3.2 estimates accident consequences for Alternatives B, C, and D.

3.1 No Action Alternative (Alternative A)—Spent Nuclear Fuel

Accidents involving spent nuclear fuel are described and analyzed in this section. First the screening results are given (Section 3.1.1), then abnormal events and design basis accidents are discussed in Section 3.1.2 and beyond design basis accidents in Section 3.1.3.

3.1.1 Screening Results for Spent Nuclear Fuel Accidents

Accidents selected as bounding for spent nuclear fuel during the screening process are listed below. Detailed screening methodology is discussed in Appendix A (Accident Screening Methodology).

Table 3.1.1-1. Spent nuclear fuel accidents.

Category[a]	Accident	Section[b]
Abnormal events	•Fuel handling accident	Tables 3.1.2-1, 3.2-1
Design basis accidents	•Fuel handling criticality •Hot Fuel Examination Facility seismic •Cask failure	3.1.2.1 3.1.2.2, Table 3.2-1 (c)
Beyond design basis accidents	•ICPP 603 seismic pool drain criticality •TAN seismic criticality •Aircraft impact	3.1.3.1, Table 3.2-1 3.1.3.2 3.1.3.3, 3.1.3.4, Table 3.2-1

a. Abnormal events are in the frequency range of 10^{-3} per year or greater. Design basis accidents are generally in the range of from 10^{-6} to 10^{-3} per year. Beyond design basis accidents are generally in the range of from 10^{-7} to 10^{-6} per year.

b. Location in this chapter where the consequence analysis or summary is located.

c. Detailed analysis not provided in this report. The accident and consequences were analyzed in existing INEL safety documentation.

3.1.2.1 Idaho Chemical Processing Plant: Inadvertent Nuclear Chain Reaction in Spent Fuel Storage Facility (1 × 10^{19} fissions 8-hour release). This section describes a reasonably foreseeable accident for irradiated fuel storage facilities at the Idaho Chemical Processing Plant (ICPP).

3.1.2.1.1 Description of Accident—An inadvertent nuclear chain reaction, associated with underwater spent nuclear fuel storage at CPP-603 was identified in the screening process as a reasonably foreseeable accident.

Standard postulated accidents identified in the literature as ones to be considered for irradiated fuel storage facilities include nuclear chain reactions, pool leaks, fuel damage events, and loss of cooling events (Elder et al. 1986, Brynda et al. 1986).

These accidents, as well as those considered in the safety analysis report (WINCO 1992a) were considered. For the following reasons, an inadvertent nuclear chain reaction was selected and other accidents eliminated:

- ICPP has previously experienced three inadvertent nuclear chain reactions. Although none of these accidents involved a fuel storage facility, they demonstrate the potential for such events.

- The consequences of water leakage from a pool draining event would have minor consequences relative to a high-level waste storage tank rupture and would thus be bounded by the high-level waste scenario (see high-level waste tank farm accident, Section 4.1.2.4). In addition, a pool leak can be part of an inadvertent nuclear chain reaction scenario.

- Mechanical fuel damage events are less impacting than a nuclear chain reaction scenario because some degree of fuel damage is typically included as part of the nuclear chain reaction scenario.

- Inadvertent nuclear chain reaction events have been addressed in virtually all DOE nonreactor environmental impact statements and safety analysis reports where these events are reasonably foreseeable, because of the perceived public concern over uncontrolled chain reactions.

Spent nuclear fuel is stored at several locations at the ICPP: (a) dry storage in the drywell vaults (CPP-749) and the Irradiated Fuels Storage Facility (CPP-603) and (b) underwater storage in CPP-666 and CPP-603. CPP-603 was selected for analysis of a nuclear chain reaction scenario for the following reasons:

Inadvertent nuclear chain reactions could result from numerous initiating events, including operator error, hanger corrosion, equipment failure, an earthquake, and aircraft crash.

Seventeen different inadvertent nuclear chain reaction scenarios were addressed in the ICPP safety document (WINCO 1992a) with a variety of initiators, events, and locations in the CPP-603 underwater storage. Additional nuclear chain reaction scenarios, which may be beyond the design basis of the facility, have also been considered in other analyses (Fast 1989). The scenario developed here for the fuel storage accident assumes a handling accident causes a criticality. Preexisting and nuclear chain reaction decay heat dissipates and thereby avoids fuel melting but still causes the release of fission products associated with 1×10^{19} fissions over an eight-hour period.

The overall likelihood for the analyzed scenario is considered to be about 1×10^{-3} per year.

3.1.2.1.2 Development of Radioactive Source Terms—The following assumptions apply to the development of the resulting source terms:

The material at risk[a] is the equivalent fission product inventory of the spent nuclear fuel involved in the nuclear chain reaction event. This inventory consists of preexisting and nuclear chain reaction-generated fission products and actinides.

(a) The preexisting inventory results from a conservative prescription of the fuel handling unit(s) that contributes to the material at risk and is a composite of multiple different fuels:

i) Enrichment: Like most of the fuel in CPP-603, the fuel is highly enriched (93.15 percent uranium-235). Lower enrichment fuel handling units are expected to be less reactive, and therefore less likely to be assembled in a critical array.

ii) Burnup: Each fuel element is assumed to have been used for 70 days at 6.25 megawatts, which is the design basis and conservative for all actual fuel.

iii) Cooling Time: The fuel has cooled only seven years since reactor operation. The last fuel receipt from the Advanced Test Reactor at CPP-603 was in the 1986-1987 period, and the majority was received years earlier.

a. Excluding stored fuel outside the critical configuration from the material at risk is a qualitative application of the damage ratio defined in Section 2.1.2.4.

Table 3.1.2.1-1. Total release fractions for inadvertent nuclear chain reaction in Spent Fuel Storage Facility accident.[a]

Material	Airborne release fraction	Respirable fraction	Leak path factor	Total
Noble gases	6.3E-01	1.0	1.0	1.0
Volatiles (H-3)	1E-05	1.0	1.0	1.0
Halogens (release to air)	6.0E-05	1.0	1.0	2.5E-01
Ruthenium	2.4E-06	1.0	1.0	1.0E-03
Nonvolatile solids	1.0E-07	1.0	1.0	5.0E-04

a. Source: Enyeart (1993).

Table 3.1.2.1-2. Total radiological source terms for inadvertent nuclear chain reaction scenario in Spent Fuel Storage Facility.[a]

Nuclide	Source term (Ci) 1×10^{19} fissions (8-h release)
Hydrogen-3 (tritium)	2.57E+01
Krypton-87	5.34E+02
Krypton-88	3.77E+02
Rubidium-88	4.62E+01
Krypton-89	1.74E+02
Rubidium-89	3.59E+01
Cesium-137	1.21E-01
Xenon-138	2.76E+03
Cesium-138	2.07E+02

a. Radionuclides listed are those that contribute 1.0E-03 (0.001) rem or more to the total dose to a maximally exposed individual at the nearest site boundary.

Table 3.1.2.1-3. Meteorological/dispersion parameters used in dosimetry calculations for inadvertent nuclear chain reaction accident at Spent Fuel Storage Facility.

Parameter	Facility worker	Nearest public access (U.S. 20/26)	Nearest site boundary[b]
Receptor distance (m)	100	5,870[a]	14,000
Release duration (h) 1×10^{19} fissions (8-h release)	8	8	8
Release height (plume rise) (m) 1×10^{19} fissions (8-h release)	0	0	0

To convert from meters to feet, multiply by 3.28.

a. Source: Section 2.1.2.7 (Table 2.1-2).

b. Nearest site boundary values (other than receptor distance) also used in calculations of doses beyond nearest site boundary.

Given the accident has occurred, the likelihood that a facility worker would experience a health effect from the 8-hour 1×10^{19} fission release in the 50 years following is approximately one in 18,500. For a member of the public, the likelihood ranges from less than one in 1,000,000 at the nearest public access down to about one in 13 million for a location about 80 kilometers from the accident. By selecting the east compass sector as the worst case in the population for both the 95 percent and 50 percent meteorological conditions (a maximum sector population of 18,000 within an 80-kilometer radius of the accident), less than one health effect is calculated to occur over the lifetime of the exposed population.

3.1.2.1.4 Preventive and Mitigative Measures—Preventive measures against inadvertent nuclear chain reactions at CPP-603 include the following:

- Thirty-centimeter (12-inch)-thick concrete dividers separate adjacent monorail rows and isolate fuel in adjacent rows in the North and Middle basins.

- The monorail hanger design incorporates bumpers at the track level and on the hanger shaft which provide at least an 46-centimeter (18-inch) center-to-center spacing between adjacent fuel storage positions in the same monorail row (provided that compatible hangers are used).

- The spent nuclear fuel storage racks are designed to be geometrically favorable for prevention of inadvertent nuclear chain reactions.

- There are administrative controls on the quantity and type of fuel placed in each storage position or on each hanger.

CPP-603 contains a Criticality Alarm System to facilitate evacuation of CPP-603 facility workers. Protection equipment (respirators, anti-Cs, etc.) are provided for the workers. CPP-603 does not have a forced ventilation system, so releases would not be released through a filtration system or stack.

3.1.2.2 Argonne National Laboratory-West: Earthquake-Induced Breach and Fuel Melt at Hot Fuel Examination Facility. The response of the Hot Fuel Examination Facility at Argonne National Laboratory-West (ANL-W) to a large seismic event was investigated. This facility is described in Section 3.1.3.3, where the radiological consequences of an aircraft crash into the facility are discussed.

3.1.2.2.1 Description of Accident—The earthquake event used for the seismic event has a return period of 100,000 years, which is beyond the design basis of the facility. A preliminary assessment of the seismic integrity of the Hot Fuel Examination Facility (Johnson 1993a) indicates that, given the current state of analysis, significant failures may be assumed to result at the Hot Fuel Examination Facility from this earthquake.

The postulated events leading to atmospheric release of radionuclides and toxic chemicals are as follows:

- The earthquake results in a peak horizontal acceleration beyond the design basis of the facility.

- This acceleration causes structural damage to the building structure and a large breach in the Main Cell.

- Coincident with the breach, failure of the fuel subassembly cooling system causes melting of fresh subassemblies.

- A bin of Waste Isolation Pilot Plant (WIPP) transuranic waste breaks open.

- Radionuclides and toxic chemicals from the WIPP transuranic waste and radionuclides from the melting fuel subassemblies are released to the atmosphere.

Table 3.1.2.2-1. Radiological material at risk for earthquake at the Hot Fuel Examination Facility.

Material	Damage ratio	Involved material	Curies
Fresh fuel (20 subassemblies)[a]	1.0	20 subassemblies[c]	Table 3.1.2.2-2
Krypton-85[b]	1.0	4 subassembly equivalents	223
Xenon-133[b]	1.0	One subassembly equivalent	6,537
Iodine-131[b]	1.0	One fuel element equivalent	140
Iodine-132[b]	1.0	One fuel element equivalent	28.3
Waste Isolation Pilot Plant transuranic Waste (one bin; 6 drums)	0.5	3 drums[c]	Table 3.1.2.2-3

a. Main Cell forced-cooling storage; 15-day decay since reactor shutdown.

b. Accumulated in Main Cell atmosphere (Adams et al. 1975).

c. Involved material = material at risk × damage ratio

3.1.2.2.3 Exposure Calculations and Results—The RSAC-5 model (Wenzel 1993) for radiological calculations and EPIcode™ (Homann 1988) for toxicological calculations, referenced here, are described in more detail in Sections 2.1.2 and 2.2.2, respectively. These sections also document the generic assumptions used in the accident assessments. The discussion below is limited to the unique aspects of the model and specific results for this accident—radiological and toxic chemical releases due to the maximum earthquake at the Hot Fuel Examination Facility.

For this scenario, the RSAC-5 program was used to determine the dose from external and internal pathways at two receptor locations: (1) a location within the INEL controlled access zones (inside the ANL-W facility area) and (2) the nearest public access and nearest site boundary at U.S. Highway 20, generally south of ANL-W. At the nearest public access/nearest site boundary location, doses are calculated both for the stranded motorist and for the individual living at nearest site boundary. The assumptions for the receptors are the same as most other accidents and as summarized in Section 2.1.2.7. Nearest site boundary assumptions and parameters were used in RSAC-5 to calculate individual doses beyond the nearest site boundary.

Table 3.1.2.2-3. Radiological source term for earthquake at the Hot Fuel Examination Facility—Waste Isolation Pilot Plant transuranic waste.

Constituent	Involved material (3 Drums) (Ci)	Airborne release fraction[a]	Respirable fraction	Leak path factor	Source term (Ci)
Plutonium-238	7.71E-01	1.0E-03	5.0E-02	1.0E+00	3.86E-05
Plutonium-239	4.20E-01	1.0E-03	5.0E-02	1.0E+00	2.10E-05
Plutonium-240	1.03E-01	1.0E-03	5.0E-02	1.0E+00	5.15E-06
Plutonium-241	3.24E+00	1.0E-03	5.0E-02	1.0E+00	1.62E-04
Americium-241	2.78E+00	1.0E-03	5.0E-02	1.0E+00	1.39E-04

a. The fraction of material that can become suspended in a spill/impact type of accident is 0.1 percent (DOE 1991c).

b. Data from DOE (1991c).

c. The earthquake scenario involves major failure of both primary and secondary confinement barriers. A leak path factor of 1.0 is assigned for a major failure of confinement barriers.

The total release duration is 30 days. This release, however, occurs in three stages (Adams et al. 1975). In the first 2 hours, 11 percent of the available noble gases and iodines are released. An additional 10 percent of the cell atmosphere is assumed to be released in the 30 days following the initiation of the event. Fission products and plutonium start escaping at the beginning of the second hour. The meteorological/dispersion parameters for RSAC-5 are as described in Section 2.1.2.5 and summarized in Table 2.1-20 (Section 2.1.4). For EPIcode™, these parameters are given in Table 2.1-3. The biological parameters for RSAC-5 are as described in Section 2.1.2.6, and exposure times do not deviate from Table 2.1-21. Scenario-specific input parameters for RSAC-5 and EPIcode™ are given in Table 3.1.2.2-6.

The radiological dose results for each applicable pathway and for their total at each of the three receptors are given in Table 3.1.2.2-7. To prevent exposure to doses greater than 5 rem to the public, intervention by evacuation or prevention of contaminated food consumption is assumed to have occurred. Dose results are also presented for various communities within 80 kilometers (50 miles) of ANL-W. The table also includes the dose for the population within an 80-kilometer radius, as calculated by the methodology described in Section 2.1.2.8.

Use of the risk factors from Section 2.1.2.9 (Table 2.1-3) with the doses calculated for this contribution to the multiple-facility event results in the calculated health effects shown in the last two columns of Table 3.1.2.2-7.

Table 3.1.2.2-5. Nonradiological source term for earthquake at the Hot Fuel Examination Facility—Waste Isolation Pilot Plant transuranic waste.

Chemical	Material at risk (3 Drums) (mg)	Airborne release fraction	Respirable fraction	Leak path factor	Source term (mg)
		Volatile Organic Compounds			
1,1,1-trichloroethane	2.52E+06	1.0E+00[a]	1.0E+00[a]	1.0E+00[d]	2.52E+06
Carbon tetrachloride	2.72E+06	1.0E+00	1.0E+00	1.0E+00	2.72E+06
1,1,2-trichloro-1,2,2-trifluoroethane	1.61E+06	1.0E+00	1.0E+00	1.0E+00	1.61E+06
Trichloroethylene	1.70E+06	1.0E+00	1.0E+00	1.0E+00	1.70E+06
Methylene chloride	1.73E+05	1.0E+00	1.0E+00	1.0E+00	1.73E+05
Methyl alcohol	3.45E+03	1.0E+00	1.0E+00	1.0E+00	3.45E+03
Butyl alcohol	1.30E+03	1.0E+00	1.0E+00	1.0E+00	1.30E+03
Xylene	8.67E+03	1.0E+00	1.0E+00	1.0E+00	8.67E+03
		Solids (Particulates)			
Cadmium	1.30E+03	1.0E-03[b]	5.0E-02[c]	1.0E+00[d]	6.50E-2
Lead	3.57E+06	1.0E-03	5.0E-02	1.0E+00	1.79E+02
Mercury	1.53E+06	1.0E-03	5.0E-02	1.0E+00	7.65E+01
Beryllium	4.77E+04	1.0E-03	5.0E-02	1.0E+00	2.39E+00
Asbestos	1.19E+06	1.0E-03	5.0E-02	1.0E+00	5.95E+01
Lithium	7.68E+05	1.0E-03	5.0E-02	1.0E+00	3.84E+01
		Other			
Nitric acid	8.22E+05	1.0E-03[b]	5.0E-02[c]	1.0E+00[d]	4.11E+02
Nitrates	1.60E+05	1.0E-03	5.0E-02	1.0E+00	8.00E+00

a. Volatile organic compounds are 100 percent airborne and respirable.

b. The fraction of material that can become suspended in a spill/impact type of accident is 0.1 percent (DOE 1991).

c. Data from DOE (1991).

d. The earthquake scenario involves major failure of both primary and secondary confinement barriers. A leak path factor of 1.0 is assigned for a major failure of confinement barriers.

Table 3.1.2.2-7. Summary of dose calculation results for earthquake at the ANL-W Hot Fuel Examination Facility at Argonne National Laboratory-West.

Receptor location	Inhalation CEDE[a] (rem)	Air immersion EDE[a] (rem)	Ground surface EDE (rem)	Ingestion CEDE (rem)	Total EDE (rem)	Likelihood of fatal cancer	Total likelihood of health effect
Facility worker (100 m)	4.6E-01	7.5E-03	1.6E-01	NA	6.2E-01	2.5E-04	3.5E-04
U.S. 20 (5,240 m)	6.4E-01	9.2E-03	3.4E-03	NA	6.5E-01	3.2E-04	4.7E-04
Nearest site boundary (5,240 m)	1.9E+00	1.1E−02	1.7E-02[b]	3.9E+00[b]	5.0E+00	2.5E-03	3.7E-03
Dose to maximally exposed individual at nearby communities (rem)							
Atomic City (21 km)	2.3E-01	2.2E-03	1.1E-01[b]	4.7E+00[b]	5.0E+00	2.5E-03	3.7E-03
Mud Lake/ Terreton (32 km)	1.6E-01	1.3E-03	1.1E-01[b]	4.7E+00[b]	5.0E+00	2.5E-03	3.7E-03
Howe (35 km)	1.4E-01	1.2E-03	1.1E-01[b]	4.8E+00[b]	5.0E+00	2.5E-03	3.7E-03
Idaho Falls/ Blackfoot (50 km)	1.1E-01	8.4E-04	8.0E-02	3.5E+00	3.7E+00	1.9E-03	2.7E-03
Arco (52 km)	1.0E-01	8.1E-04	7.7E-02	3.4E+00	3.6E+00	1.8E-03	2.6E-03
Rigby (60 km)	9.0E-02	7.0E-04	6.8E-02	3.0E+00	3.2E+00	1.6E-03	2.3E-03
Craters of the Moon National Monument (73 km)	7.7E-02	5.8E-04	5.8E-02	2.6E+00	2.7E+00	1.4E-03	2.0E-03
Rexburg (74 km)	7.6E-02	5.8E-04	5.7E-02	2.5E+00	2.6E+00	1.3E-03	1.9E-03
Dose to maximum sector population within 80-km (50-mile) radius (person-rem)						Number of fatal cancers	Total number of individuals experiencing health effects
Maximum sector population 26,000 95%	1.1E+03	8.8E+00	8.3E+02	1.2E+04	1.4E+04	7.0E+00	1.0E+01
Maximum sector population 190 50%	7.2E+00	6.4E-02	5.6E+00	7.6E+01	8.9E+01	4.5E-02	6.5E-02

a. EDE — effective dose equivalent; CEDE — committed EDE.

b. Protective actions are assumed to have taken place to limit dose to affected population to the protective action guideline of 5 rem.

The toxicological source term (see Table 3.1.2.3-3) is based on the cadmium in the electrorefining process.

Table 3.1.2.3-1. Radiological material at risk for earthquake at the Fuel Cycle Facility.

Material	Damage ratio	Involved material	Curies
Exposed fuel[a]	1.0	45 kg (12.2 subassembly equivalent)	Table 3.1.2.3-2
Krypton-85[b]	1.0	1.81E+03 Ci	1.81E+03

To convert kilograms to pounds, multiply by 2.2.

a. 90-day decay since reactor shutdown.

b. Accumulated in Main Cell atmosphere (ANL-W 1993).

3.1.2.3.3 Exposure Calculations and Results—The RSAC-5 model (Wenzel 1993) for radiological calculations and EPIcode™ (Homann 1988) for toxicological calculations, referenced here, are described in more detail in Sections 2.1.2 and 2.2.2, respectively. These sections also document the generic assumptions used in the accident assessments. The discussion below is limited to the unique aspects of the model and specific results for this accident—radiological and toxic chemical releases due to the maximum earthquake at the Fuel Cycle Facility.

For this scenario, the RSAC-5 program was used to determine the dose from external and internal pathways at three receptor locations: (1) a location within the INEL controlled access zones (inside the ANL-W facility area), (2) the nearest public access, and (3) nearest site boundary at U.S. Highway 20, generally south of ANL-W. At the nearest public access/nearest site boundary location, doses are calculated both for the stranded motorist and for the individual living at nearest site boundary. The assumptions for the receptors are the same as most other accidents and as summarized in Section 2.1.2.7. Nearest site boundary assumptions and parameters were used in RSAC-5 to calculate individual doses beyond the nearest site boundary.

Table 3.1.2.3-3. Nonradiological source term for earthquake at the Fuel Cycle Facility.

Constituent	Material at risk (kg)	Airborne release fraction[a]	Respirable fraction	Leak path factor	Source term (kg)
Cadmium	6.0E+01	1.0E-02	1.0E+00	1.0E+00	6.0E-01

To convert kilograms to pounds, multiply by 2.2.

a. Data from Elder et al. (1986).

The total release duration is 30 days. The release, however, occurs in three stages (Adams et al. 1975). In the first 2 hours, 11 percent of the available noble gases and iodines are released. An additional 10 percent of the cell atmosphere is assumed to be released in the 30 days following the initiation of the event. Fission products and plutonium start to escape at the beginning of the second hour. The meteorological/dispersion parameters for RSAC-5 are as described in Section 2.1.2.5 and summarized in Table 2.1-20. For EPIcode™, these parameters are given in Table 2.2-1. The biological parameters for RSAC-5 are as described in Section 2.1.2.6, and exposure times do not deviate from Table 2.1-21. Scenario-specific input parameters for RSAC-5 and EPIcode™ are given in Table 3.1.2.3-4.

The radiological dose results for each applicable pathway and for their total at each of the three receptors are given in Table 3.1.2.3-5. Dose results are also presented for various communities within 80 kilometers (50 miles) of ANL-W. To prevent exposures to doses greater than 5 rem to the public at the nearest site boundary, intervention by evacuation or prevention of contaminated food consumption would be required. The table also includes the dose for the population within an 80-kilometer radius, as calculated by the methodology described in Section 2.1.2.8.

Use of the risk factors from Section 2.1.2.9 (Table 2.1-3) with the doses calculated for this contribution to the multiple-facility accident results in the calculated health effects shown in the last two columns of Table 3.1.2.3-5.

The EPIcode™ calculations show that no health effects would result to the general public from the postulated toxic chemical release. Cadmium has no Emergency Response Planning Guideline (ERPG)-1 value, so airborne concentrations are compared to threshold limit value/time-weighted average (TLV/TWA) values. For facility workers at ANL-W, the calculated concentration is less than 50 percent above the TLV/TWA value of 0.05 milligrams per cubic meter. The postulated exposure of five minutes is sufficiently short and the concentration sufficiently close to TLV/TWA that no health effects would result.

Table 3.1.2.3-5. Summary of dose calculation results for earthquake at the Fuel Cycle Facility at Argonne National Laboratory-West.

Receptor location	Inhalation CEDE[a] (rem)	Air immersion EDE[a] (rem)	Ground surface EDE (rem)	Ingestion CEDE (rem)	Total EDE (rem)	Likelihood of fatal cancer	Total likelihood of health effect
Facility worker (100 m)	0.0E+00	3.2E-05	0.0E+00	NA	3.2E-05	1.3E-08	1.8E-08
U.S. 20 (5,240 m)	1.8E-01	1.1E-03	1.1E-04	NA	1.8E-01	9.0E-05	1.3E-04
Nearest site boundary (5,240 km)	2.8E-01[b]	1.6E-03[b]	5.2E-01[b]	4.2E+00[b]	5.0E+00	2.5E-03	3.7E-03
Dose to maximally exposed individual at nearby communities (rem)							
Atomic City (21 km)	1.0E-01	6.3E-04	2.6E-01	1.7E+00	2.0E+00	1.0E-03	1.5E-03
Mud Lake/Terreton (32 km)	6.7E-02	4.3E-04	1.7E-01	1.1E+00	1.3E+00	6.5E-04	9.5E-04
Howe (35 km)	6.2E-02	3.9E-04	1.6E-01	1.0E+00	1.3E+00	6.5E-04	9.5E-04
Blackfoot/Idaho Falls (50 km)	4.6E-02	2.9E-04	1.2E-01	7.6E-01	9.3E-01	4.7E-04	6.8E-04
Arco (52 km)	4.4E-02	2.8E-04	1.1E-01	7.4E−01	9.0E-01	4.5E-04	6.6E-04
Rigby (60 km)	3.9E-02	2.5E-04	1.0E-01	6.6E-01	8.0E-01	4.0E-04	5.8E-04
Craters of the Moon National Monument (73 km)	3.3E-02	2.1E-04	8.6E-02	5.6E-01	6.8E-01	3.4E-04	5.0E-04
Rexburg (74 km)	3.3E-02	2.1E-04	8.5E-02	5.5E-01	6.7E-01	3.4E-04	4.9E-04
Dose to maximum sector population within 80-km (50-mile) radius (person-rem)						Number of fatal cancers	Total number of individuals experiencing health effects
Maximum sector population 26,000 95%	4.8E+02	3.1E+00	1.2E+03	4.4E+03	6.1E+03	3.1E+00	4.5E+00
Maximum sector population 5,100 50%	3.1E+00	2.0E-02	8.2E+00	3.0E+01	4.1E+01	2.1E-02	3.0E-02

a. EDE — effective dose equivalent; CEDE — committed EDE.

b. Protective actions are assumed to have taken place to limit doses to affected population to 5 rem in accordance with protective action guidelines.

A large INEL earthquake was postulated as the initiating event because it can result in the following events:

- Loss of geometric control over the fuel due to failure of components such as monorail and rack
- Building structural failure and collapse
- Prompt loss of pool water (i.e., the wave action can remove a substantial portion of the pool water)
- Loss of pool integrity and subsequent pool draining
- Fuel cladding failure caused to mechanical damage from structural members.

This earthquake is postulated to produce a bedrock acceleration at ICPP that is beyond the design basis for the facility (the earthquake has an approximately return period of 100,000 years). The scenario includes the following major events:

- The initiating earthquake produces a critical configuration of some spent nuclear fuel in a portion of one basin.
- The earthquake also causes structural damage and failure of the CPP-603 building, which results in a complete draining of the basin.
- The preexisting and nuclear chain reaction-generated decay heat melts two Advanced Test Reactor fuel elements and causes the release of fission products in a single burst of 3×10^{19} fissions.

The overall likelihood for the set of assumptions for the analyzed scenario is considered to be in the range of 1×10^{-7} per year.

3.1.3.1.2 Development of Radioactive Source Terms—The material at risk is the equivalent fission product inventory of the spent nuclear fuel involved in the nuclear chain reaction event. This inventory consists of preexisting and nuclear chain reaction-generated fission products and actinides. The assumptions regarding the preexisting inventory listed in Section 3.1.2.1.2 also apply to the development of the source terms for this accident. However, the decay heat one hour after the nuclear chain reaction[a] is sufficient only to partially melt two fuel elements.

a. The fuel array must be fully flooded to produce an inadvertent nuclear chain reaction; however, fuel melting can result only if there is virtually no water present. Therefore, the chain reaction must occur with at

Table 3.1.3.1-1. Total release fractions for inadvertent nuclear chain reaction in Spent Fuel Storage Facility accident (3×10^{19} burst release).[a]

Material	Airborne release fraction	Respirable fraction	Leak path factor	Total
Noble gases	1.0	1.0	1.0	1.0
Volatiles (H-3)	1.0	1.0	1.0	1.0
Halogens (release to air)	2.5E-01	1.0	1.0	2.5E-01
Ruthenium	1.0E-03	1.0	1.0	1.0E-03
Nonvolatile solids	5.0E-04	1.0	1.0	5.0E-04

a. Sources: Adams and Carboneau (1992), NRC (1977b).

For this scenario, the RSAC-5 program was used to determine the dose from external and internal pathways at three receptor locations. The three receptor locations are (a) a location within the INEL controlled access zones (inside the ICPP facility area), (b) the nearest public access at U.S. Highway 20/26, and (c) the nearest site boundary, generally south of ICPP. The assumptions for the three receptors are the same as most other accidents and as summarized in Section 2.1.2.7. Nearest site boundary assumptions and parameters were used in RSAC-5 to calculate individual doses beyond the nearest site boundary.

The fission product release was modeled as an instantaneous burst. (Section 3.1.2.1 models an eight-hour linear release.)

The meteorological/dispersion parameters for RSAC-5 are as described in Section 2.1.2.5 and summarized in Table 2-20. The biological parameters for RSAC-5 are as described in Section 2.1.2.6 and exposure times do not deviate from Table 2-21. Scenario-specific input parameters for RSAC-5 are given in Table 3.1.3.1-3.

The primary risk to co-located facilities as a result of an inadvertent fuel storage nuclear chain reaction at ICPP would be caused by the required evacuation, leaving such facilities unattended. If emergency procedures are followed before evacuation, initiation of accidents at other ICPP facilities is not expected. Before resumption of normal operation at ICPP facilities, extensive ground surface decontamination of the ICPP area would be necessary.

The dose results for each applicable pathway and for their total at each of the three receptor locations are given in Table 3.1.3.1-4. Dose results are also presented for various communities within 80 kilometers (50 miles) of the ICPP. The tables also include the doses for the population within a 80-kilometer radius, as calculated by the methodology described in Section 2.1.2.8.

Table 3.1.3.1-3. Meteorological/dispersion parameters used in dosimetry calculations for inadvertent nuclear chain reaction accident at Spent Fuel Storage Facility (3×10^{19} burst release).

Parameter	Facility worker	Nearest public access (U.S. 20/26)	Nearest site boundary[b]
Receptor distance (m)	100	5,870[a]	14,000
Release duration (s)	1	1	1
Release height (plume rise) (m)	200	200	200

To convert meters to feet, multiply by 3.28.

a. Source: Section 2.1.2.7 (Table 2.1-2).

b. Nearest site boundary values (other than receptor distance) also used in calculations of doses beyond nearest site boundary.

Doses for facility workers from direct radiation due to the nuclear chain reaction were not calculated because the basin walls will provide effective shielding.

Use of the risk factors from Section 2.1.2.9 with the doses calculated for this scenario results in the calculated health effects shown in the last two columns of Table 3.1.3.1-4.

Given the accident has occurred, the likelihood that a facility worker would experience a health effect from the burst of 3×10^{19} fission (burst release) accident in the 50 years following is approximately one in 200. For a member of the public, the likelihood ranges from less than one in 27,500 at the nearest public access down to about one in 270,000 for a location about 80 kilometers (50 miles) from the accident. By selecting the east compass sector as the worst case in the population for both the 95 percent or 50 percent meteorological conditions, a maximum sector population of 18,000 within an 80-kilometer radius of the accident, less than one health effect is calculated to occur over the lifetime of the exposed population.

3.1.3.1.4 Preventive and Mitigative Measures—Preventive measures for this inadvertent nuclear chain reaction at CPP-603 are the same as those described in Section 3.1.2.1.4 for the 1×10^{19} 8-hour release.

3.1.3.2 Test Area North Hot Cell Complex: Inadvertent Nuclear Chain Reaction in Spent Fuel. This section describes the bounding reasonably foreseeable accident for irradiated fuel storage facilities at the Test Area North (TAN).

3.1.3.2.1 Description of Accident—An inadvertent nuclear chain reaction, associated with spent nuclear fuel[a] storage at TAN and initiated by an earthquake, was identified in the preliminary screening process as the bounding reasonably foreseeable accident.

Standard postulated accidents identified in the literature as ones to be considered for irradiated fuel storage facilities include inadvertent nuclear chain reactions, pool leaks, fuel damage events, and loss of cooling events (Elder et al. 1986, Brynda et al. 1986).

These accidents, as well as those of the TAN Hot Cell Complex Final Safety Analysis Report (EG&G Idaho 1991) and of the Preliminary Hazard Assessment for EG&G Waste Management Facilities (EG&G Idaho 1992a) were considered. A qualitative process of comparing the volatile radionuclide inventories and release fractions for each candidate scenario was used to identify a small number of scenarios that bound the other scenarios. A fuel damage and inadvertent nuclear chain reaction accident was selected and other accidents eliminated because of the following considerations:

- INEL has experienced three inadvertent nuclear chain reactions, all at the ICPP. Though none of these accidents involved a spent nuclear fuel storage facility (and none has occurred nationally (Knief 1991)), they demonstrate the potential for such accidents.

- The consequences of a water leakage from a storage pool draining event would have relatively minor consequences by itself[b] because the spent nuclear fuel has decayed sufficiently that it would not melt. A pool leak can be part of an overall inadvertent nuclear chain reaction scenario.

- Fuel damage scenarios are required for safety analysis report consideration by Regulatory Guide 1.25 (NRC 1972). A precedent exists for consideration of damage to the entire fuel contents of a cask.

- Inadvertent nuclear chain reactions have been addressed in virtually all DOE nonreactor environmental impact statements and safety analysis reports where the reactions are reasonably foreseeable, due in part to the perceived public concern over uncontrolled chain reactions.

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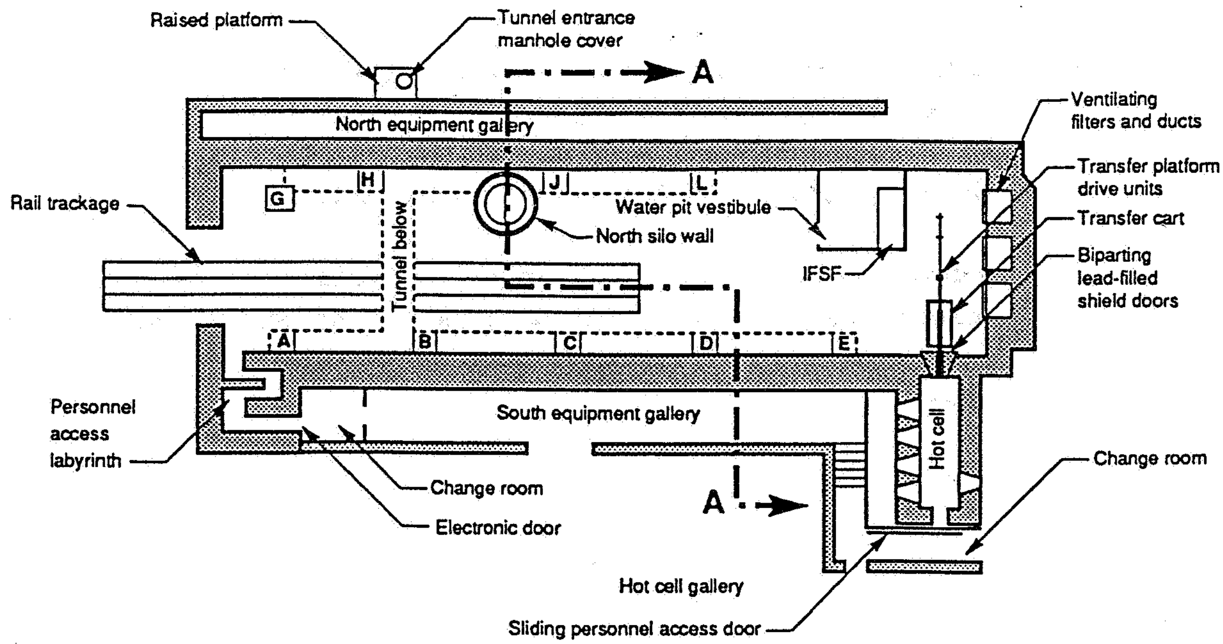
a. The term spent nuclear fuel is used here to include the Three Mile Island-2 core debris and cuno filters, even though they are not strictly spent nuclear fuel.

b. Releases of radioactive materials from commercial spent nuclear fuel will occur only if temperatures are high enough (850°C to 950°C) to cause cladding rupture (Benjamin et al. 1979). Commercial spent nuclear fuel in a drained pool with inadequate or inoperative ventilation is safe for minimum decay times of two to four years (Benjamin et al. 1979). All spent nuclear fuel at Test Area North has had considerably greater decay times.



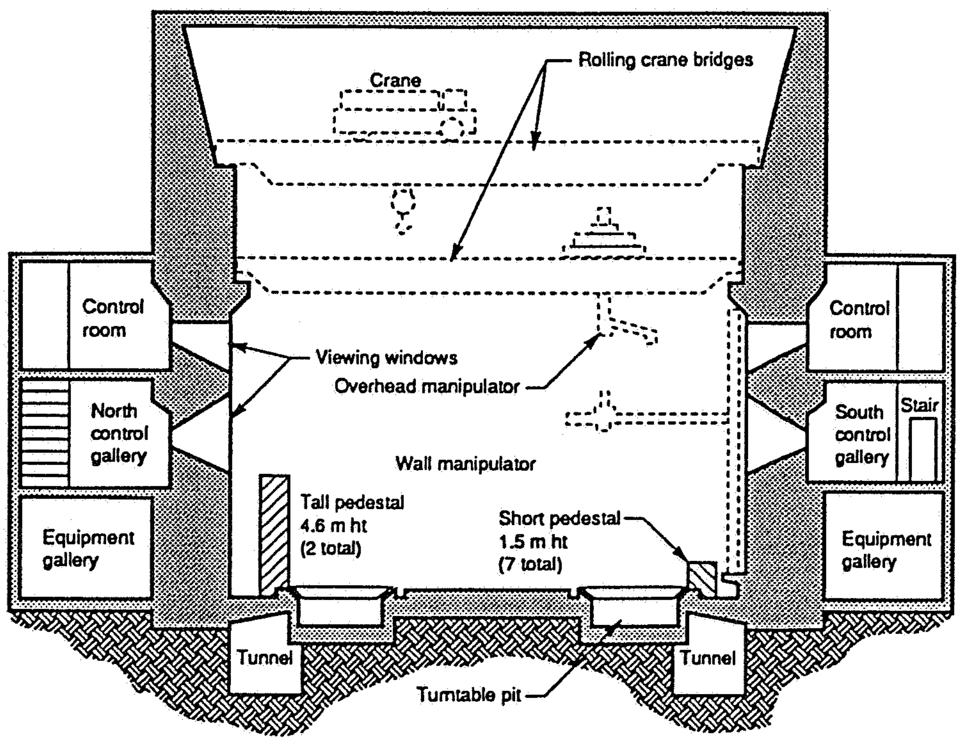


Figure 3.1.3.2-1. Illustration of first floor and elevation view of the Test Area North Hot Shop.

- The earthquake results in failure of the Hot Shop roof and doors.

- Both the V-21 and MC-10 casks are in the Hot Shop with their lids off or at least unsecured.

- Either the falling debris or the seismic accelerations result in the toppling of casks.

- Toppling of casks with the lid removed results in the spilling of the entire contents of the casks.

- The spilled spent nuclear fuel forms a critical array because of the significant quantity of material (21 fuel assemblies in the V-21 cask and 18 assemblies in the MC-10 cask) with no geometric control and because of the potential for water moderation.

On the basis of the conservative conditional probabilities for the major events discussed above, the overall scenario frequency is in the range of 1×10^{-6} to 1×10^{-7} per year.

3.1.3.2.2 Development of Radioactive Source Term—The material at risk is the equivalent fission product inventory of the spent nuclear fuel involved in the inadvertent nuclear chain reaction event. This inventory consists of preexisting and nuclear chain reaction-generated fission products and actinides, and of crud on the spent nuclear fuel cladding. The following assumptions apply to the development of the resulting source term:

(a) The preexisting inventory results from a conservative prescription of the fuel types that contribute to the material at risk.

  (i) Number of Assemblies: 39 fuel assemblies, the combined contents of the General Nuclear Services Inc. V-21 Cask (GNS 1985) and the Westinghouse MC-10 cask (WNES 1984) (the two casks with the largest capacity at Test Area North) are involved.

  (ii) Fuel Type: The fuel in the casks is commercial reactor spent nuclear fuel. Commercial pressurized water reactor (PWR) spent nuclear fuel provides the bounding case for this analysis since commercial boiling water reactor (BWR) spent nuclear fuel has a lower burnup (Funk and Jacobson 1979).

  (iii) Burnup: The involved spent nuclear fuel has a total uranium burnup of 35,170 megawatt days per metric ton uranium (MWd/MTU) (GNS 1985), slightly greater than the maximum typical pressurized water reactor burnup of 33,160 Mwd/MTU (Funk and Jacobson 1979). This burnup is consistent with historical records that indicate significant amounts of high burnup fuel were

Table 3.1.3.2-1. Summary of airborne release fractions and respirable fractions for inadvertent nuclear chain reaction accident at Hot Cell Complex at Test Area North.[a]

Radionuclide group	Airborne release fraction: 1×10^{19} fissions (8-h release)	Airborne release fraction: 3×10^{19} fissions (burst release)	Respirable fraction
Krypton-85	0.3	0.3	1.0
Other noble gases	0.63	1.0	1.0
Volatiles (e.g., H-3)	1E-05	1.0	1.0
Ruthenium	2.4E-06	1E-03	1.0
Halogens	6.0E-05	2.5E-02	1.0
Nonvolatile solids	1E-07	5E-04	1.0
Crud release	1.0	1.0	0.001[b]

a. Sources: NRC (1972); Enyeart (1993).

b. Source: Einziger and Cook (1984).

(f) The respirable fraction is assumed to be 0.1 percent (Einziger and Cook 1984) for the crud. It is conservatively set at 100 percent for the other nonvolatile solids because particle size is assumed to be 1.0 micron aerodynamic diameter.

(g) No credit is taken for any reduction and removal such as plate-out scrubbing or filtration.

(h) Since the TAN-607 superstructure is assumed to fail, 100 percent of the release to the building atmosphere is also released to the environment, so the leak path factor is 1.0. The burst of 3×10^{19} fissions assumes a plume rise due to the violent nature of the reaction. The eight-hour 1×10^{19} fission release neglects plume rise.

The source term resulting from combining the material at risk with the above assumptions on airborne and respirable fractions and leak path factor is summarized in Table 3.1.3.2-2. It is conservative for virtually every individual parameter and bound the source term for all other reasonably foreseeable scenarios. The considerable uncertainty associated with the scenario is bounded by the conservatisms in the assumptions as discussed below:

- An earthquake of this or any other magnitude will not necessarily cause an inadvertent nuclear chain reaction. No inadvertent nuclear chain reaction event has occurred anywhere associated with spent nuclear fuel storage (Knief 1991).

- Compared to the assumed criticality fission yield of 3×10^{19}, only one out-of-core criticality event has produced greater than 3×10^{18} total fissions (Knief 1991). It produced about 4×10^{19} total fissions during 20 minutes of critical pulses in a uranium-bearing solution. The initial critical pulse was estimated to be about 1×10^{17} fissions.

- 100 percent fuel cladding failure is unlikely because all of the assemblies would probably not spill from the casks due to the gridded storage section. A cask will provide considerable protection for any assembly remaining within.

- Most of the spent nuclear fuel has cooled for considerably more than 10 years, some as long as 22 years. The mean decay time for the fuel during the scope of the EIS (DOE 1995) (through Fiscal Year 2005) is roughly 20 years. Spent nuclear fuel activity is reduced by about 30 percent from 4.3×10^5 curies at 10 years to 3.1×10^5 curies at 20 years (GNS 1985).

- The release fractions are more conservative than those in NRC guidance (NRC 1972). Volatile fission products (H-3) are treated the same as noble gases. The NRC guidance does not account for release of nonvolatile fission/activation products which are included here.

3.1.3.2.3 Dose Calculations and Results—The RSAC-5 (Wenzel 1993) and ORIGEN 2.1 (Croff 1983) models referenced here are described in more detail in Section 2.1.2. That section documents generic assumptions and parameters used in the accident assessment. The discussion below is limited to the unique aspects of the model and specific results for this bounding accident—inadvertent nuclear chain reaction in the Hot Shop Complex at Test Area North.

RSAC-5 fission product inventory calculations for reactor operation were based on 39 fuel assemblies with 465 kilograms (1,024 pounds) uranium per assembly, with the previously specified fuel burnup of about 35,000 megawatt days per metric ton of uranium. This fission product inventory was decayed for 10 years and then compared to the ORIGEN estimates for 10-year cooled fuel (see GNS 1985) as an independent verification. For nuclides important to dose calculations, i.e., those with half-lives greater than about ten seconds, the two outputs were within about two percent.

Table 3.1.3.2-3. Meteorological/dispersion parameters used in dosimetry calculations for inadvertent nuclear chain reaction at TAN Hot Cell Complex.

Parameter	Facility worker	Nearest public access (State Route 33)	Nearest site boundary[a]
Receptor distance (m)	100	864[b]	9,170
Release duration			
1×10^{19} fissions	8 h	8 h	8 h
3×10^{19} fissions	1 s	1 s	1 s
Release height (plume rise) (m)			
1×10^{19} fissions	0	0	0
3×10^{19} fissions	180	180	180

To convert meters to feet, multiply by 3.28.

a. Nearest site boundary values (other than receptor distance) also used in calculations of doses beyond nearest site boundary.

b. Source: Section 2.1.2.7 (Table 2.1-2).

The dose results for each applicable pathway and for their total at each of the three receptor locations are given in Tables 3.1.3.2-4 and 3.1.3.2-5. Dose results are also presented for various communities within 80-kilometer (50 miles) of the Test Area North facility area. The tables also includes the dose for the population with a 80-kilometer radius, as calculated by the methodology described in Section 2.1.2.8.

Use of the risk factors from Section 2.1.2.9 with the doses calculated for this scenario results in the calculated health effects shown in the last two columns of Tables 3.1.3.2-4 and -5. The likelihood that a facility worker would experience a health effect from the burst of 3×10^{19} in the 50 years following the accident is approximately one in 10. For a member of the public, the likelihood ranges from less than one in 80 at the nearest site boundary down to about one in 500 for a location about 80 kilometers (50 miles) from the accident. By selecting the east compass sector as the worst case in the population for both the 95 percent or 50 percent meteorological conditions (a maximum sector population of 18,000 within an 80-kilometer radius of the accident), six health effects are calculated to occur over the lifetime of the exposed population.

The likelihood that a facility worker would experience a health effect from the 1×10^{19} fissions over an 8-hour release in the 50 years following the accident is approximately one in 370,000. For a member of the public, the likelihood ranges from less than one in 4,800 at the nearest site boundary down to about one in 31,000 for a location about 80 kilometers from the accident. By selecting the east compass sector as the worst case in the population for both the 95

Table 3.1.3.2-5. Summary of dose calculation results for inadvertent nuclear chain reaction in spent fuel at Hot Cell Complex at Test Area North for the 1×10^{19} fission scenario (8-hour release).

Receptor location	Inhalation CEDE[a] (rem)	Air immersion EDE[a] (rem)	Ground surface EDE (rem)	Ingestion CEDE (rem)	Total EDE (rem)	Likelihood of fatal cancer	Total likelihood of health effect
Facility worker (100 m)	2.5E-03	2.2E-03	1.5E-04	NA	4.9E-03	2.0E-06	2.7E-06
Idaho 33 (864 m)	9.5E-02	7.4E-02	1.2E-03	NA	1.6E-01	8.0E-05	1.2E-04
Nearest site boundary (9170 m)	2.1E-02	1.4E-02	1.4E-01	1.1E-01	2.9E-01	1.5E-04	2.1E-04
Dose to maximally exposed individual at nearby communities (rem)							
Mud Lake/Terreton (14 km)	3.5E-02	9.0E-03	9.1E-02	7.3E-02	2.1E-01	1.1E-04	1.5E-04
Howe (30 km)	1.7E-02	4.2E-03	4.4E-02	3.5E-02	1.0E-01	5.0E-05	7.3E-05
Atomic City (47 km)	1.1E-02	2.8E-03	2.9E-02	2.3E-02	6.6E-02	3.3E-05	4.8E-05
Arco (58 km)	9.4E-03	2.3E-03	2.4E-02	2.0E-02	5.5E-02	2.8E-05	4.0E-05
Idaho Falls (60 km)	9.1E-03	2.2E-03	2.4E-02	1.9E-02	5.4E-02	2.7E-05	3.9E-05
Rigby (62 km)	8.9E-03	2.2E-03	2.3E-02	1.9E-02	5.3E-02	2.7E-05	3.9E-05
Rexburg (74 km)	7.7E-03	1.9E-03	2.0E-02	1.6E-02	4.6E-02	2.3E-05	3.4E-05
Blackfoot (77 km)	7.5E-03	1.8E-03	1.9E-02	1.6E-02	4.4E-02	2.2E-05	3.2E-05
Dose to maximum sector population within 80-km (50-mile) radius (person-rem)						Number of fatal cancers	Total number of individuals experiencing health effects
Maximum sector population 8,300 95%	7.2E+01	1.8E+01	1.8E+02	6.8E+01	3.4E+02	1.7E-01	2.5E-01
Maximum sector population 18,000 50%	2.1E+00	5.0E-01	5.5E+00	2.0E+00	1.0E+01	5.0E-03	7.3E-03

a. EDE — effective dose equivalent; CEDE — committed EDE.

3.1.3.3 Argonne National Laboratory-West: Aircraft Crash into Hot Fuel Examination Facility (Radiological). This section describes the radiological consequences of the bounding reasonably foreseeable accident at the Argonne National Laboratory-West (ANL-W) Hot Fuel Examination Facility (HFEF).[a]

3.1.3.3.1 Description of the Accident—A crash of a large commercial jet transport into the HFEF was selected as the bounding reasonably foreseeable accident. The accident initiated by this event results in penetration of the Main Cell and in a fire in the facility involving aviation fuel.

DOE guidance (Brynda et al. 1986) on accident selection both for Irradiated Fissile Material Storage Facilities and for Hot Laboratories applies to the HFEF. Postulated accidents and initiating events to be considered include nuclear chain reactions; fires and explosions; severe natural phenomena; external manmade hazards; fuel element failures; and failure of process handling, cooling, or shielding equipment. These accidents and initiating events were considered in the evaluation summarized below.

Potential accident scenarios, both internally and externally initiated, were compared for this analysis with accidents in the safety analysis reports for the HFEF (Adams et al. 1975) and HFEF/S[b] (ANL-W 1988) and in the environmental impact statement for the Waste Isolation Pilot Plant (DOE 1990b).

Of the externally initiated events, an aircraft crash into the HFEF is considered to be the bounding initiating event, because it could (a) cause a major breach of confinement barriers, (b) involve a large portion of the material at risk, and (c) have a high-energy release mechanism (physical impact followed by sustained fire).

This accident was selected and other accidents eliminated because reasonably foreseeable internally initiated design basis-type events do not have sufficient energy sources to cause a significant breach of confinement and release to the atmosphere. The facility contains little combustible material to sustain a fire, and the primary material at risk in the facility is inside the Main Cell, which has an argon atmosphere. An inadvertent nuclear chain reaction is extremely unlikely but credible. However, occurrence of an inadvertent nuclear chain reaction coincident with failure of cell barriers and filtration is not reasonably foreseeable.

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a. The toxicological consequences of this accident are discussed in Section 7.1.3.2. The consequences (both radiological and toxicological) of the maximum seismic event at the Hot Fuel Examination Facility (HFEF) at Argonne National Laboratory-West are discussed in Section 3.1.2.2.

b. Accident analyses for HFEF/S, now known the Fuel Cycle Facility, are presented in Section 3.1.3.4.

- The ensuing fire involves the contents of the Main Cell and Waste Isolation Plant transuranic waste in the Decontamination Cell, High Bay Area, and Hot Repair Area, resulting in atmospheric release of radionuclides.

For determining aircraft crash probability, the bounding scenario is limited to large or high-velocity jet airplanes. High-velocity military jets from the U.S. Air Force base at Mountain Home in southwestern Idaho could possibly enter the airspace of the INEL. Also, large jet aircraft (B-747, C-5A) have been flown at low altitude over portions of the INEL for vortex tests. The likelihood of a large aircraft crash into the HFEF is very remote, but possible. Analyses of jet aircraft crashes at specific facilities have resulted in predicted frequencies on the order of 1×10^{-7} to 1×10^{-8} per facility year (Lee et al. 1994, NRC 1975). Therefore, based on the likelihood of the aircraft crash alone, the ANL-W/HFEF material release accident is estimated to be 1×10^{-7} per year.

3.1.3.3.2 Development of Radioactive Source Term—The material at risk during the aircraft crash scenario, and the material involved in the scenario (material at risk × damage ratio as discussed in Section 2.1.2.4), are summarized in Table 3.1.3.3-1, and supported by Tables 3.1.3.3-2 through 3.1.3.3-4. This material consists of reactor fuel and fission product gases in the Main Cell and Waste Isolation Pilot Plant transuranic waste in various locations. The assumptions necessary to assure that the resulting source term is reasonably bounding for the postulated accident scenario are discussed below.

- Forty storage locations in the floor of the Main Cell are designed to provide for forced cooling of fresh fuel. Storage locations for 50 other subassemblies are above the floor of the Main Cell; the total inventory of fuel subassemblies in the Main Cell ranges between 100 and 150.

- Half of the fuel subassemblies stored above the floor are mechanically damaged in the impact and are involved in the ensuing fire (a total of 25). This damage ratio is reasonable given the storage configuration of the fuel and postulated damage to the cell in the accident. The fuel cooling time of 150 days is a conservative median for fuel not requiring forced cooling in the cell.

- The current maximum credible accident for the safety analysis report is based on 33 fuel subassemblies with 15-day cooling in the Argon Cell. However, this inventory is not reasonably foreseeable given the current mission of HFEF. For this analysis, twenty fuel assemblies that require forced cooling are assumed in the cell during the accident. Although this inventory is half of the forced cooled capacity, it is an extremely conservative upper bound for foreseeable operating conditions. Failure of the forced cooling in the accident results in melting of the 20 fuel subassemblies. Fifteen-day cooling is a conservative cooling time for the 20 subassemblies (considering restrictions on removing fuel from the reactor and moving it to HFEF).

Table 3.1.3.3-2. Plutonium isotopic mass in one subassembly in main cell at Hot Fuel Examination Facility[a]

Plutonium isotope	Mass (g)	Weight fraction	Curies
Plutonium-238	1.97E-01	3.0E-04	3.43E+00
Plutonium-239	5.81E+02	8.84E-01	3.57E+01
Plutonium-240	6.83E+01	1.04E-01	1.56E+01
Plutonium-241	8.54E+00	1.3E-02	9.58E+02
Plutonium-242	2.63E-01	4.0E-04	1.03E-03

To convert grams to pounds, multiply by 2.2×10^{-3}.

a. Data from Adams et al. (1975).

Table 3.1.3.3-3. Fifteen-day fission product inventory in one fuel subassembly at main cell at Hot Fuel Examination Facility[a]

Nuclide	Source term (Ci)	Nuclide	Source term (Ci)
Krypton-85	5.57E+01	Tellurium-132	1.02E+03
Strontium-89	3.57E+02	Iodine-131	5.17E+03
Strontium-90	3.95E+02	Iodine-132	1.05E+03
Yttrium-90	3.95E+02	Xenon-133	6.54E+03
Yttrium-91	1.77E+04	Cesium-134	9.40E+01
Zirconium-95	2.45E+04	Cesium-136	1.08E+03
Niobium-95	2.86E+04	Cesium-137	5.76E+02
Molybdenum-99	7.90E+02	Barium-140	1.36E+04
Technetium-99m	7.90E+02	Lanthanum-140	1.63E+04
Ruthenium-103	2.18E+04	Cerium-141	2.35E+04
Ruthenium-106	3.68E+03	Cerium-144	8.89E+03
Rhodium-103m	2.18E+04	Praseodymium-143	1.36E+04
Rhodium-106	3.68E+03	Praseodymium-144	8.89E+03
Silver-111	3.12E+02	Neodymium-147	4.67E+03
Antimony-125	1.24E+02	Promethium-147	1.88E+03
Tellurium-127m	1.81E+02	Europium-155	1.61E+02
Tellurium-127	5.17E+02		

a. Data from Adams et al. (1975).

The source term is developed by multiplying the appropriate quantities summarized in Table 3.1.3.3-1 by the airborne release fractions, respirable fractions, and leak path factors summarized in Table 3.1.3.3-5. Total resulting source terms for the aircraft crash accident are specified in Table 3.1.3.3-6. The values reported in the tables reflect the material that is involved in the accident, i.e., 23 drums of transuranic waste, 45 fuel subassemblies (20 15-day cooled and 25 150-day cooled), and radionuclides in the Main Cell atmosphere at the time of the accident (krypton, xenon, iodine).

3.1.3.3.3 Dose Calculations and Results—The RSAC-5 (Wenzel 1993) model referenced here is described in more detail in Section 2.1.2. That section also documents the generic assumptions used in the accident assessments. The discussion below is limited to the unique aspects of the model and specific results for this bounding accident—radiological release due to a large aircraft crash into ANL-W HFEF.

For this scenario, the RSAC-5 program was used to determine the dose from external and internal pathways at three receptor locations. These receptor locations are (1) a location within the INEL controlled access zones (inside the ANL-W facility area), (2) the nearest public access, and (3) the nearest site boundary at U.S. Highway 20, generally south of ANL-W. At the nearest public access/nearest site boundary location, doses are calculated both for the stranded motorist and for an assumed individual living at the nearest site boundary. The assumptions for the receptors are the same as most other accidents and as summarized in Section 2.1.2.7. Nearest site boundary assumptions and parameters were used in RSAC-5 to calculate individual doses beyond the nearest site boundary.

The release duration is 60 minutes. The meteorological/dispersion parameters for RSAC-5 are described in Section 2.1.2.5 and summarized in Table 2.1-20. The biological parameters for RSAC-5 are described in Section 2.1.2.6, and exposure times do not deviate from Table 2.1-21. Scenario-specific input parameters for RSAC-5 are given in Table 3.1.3.3-7.

The dose results for each applicable pathway and for their total at each of the three receptors are given in Table 3.1.3.3-8. Dose results are also presented for various communities within 80 kilometers (50 miles) of ANL-W. The table also includes the dose for the population within an 80-kilometer radius, as calculated by the methodology described in Section 2.1.2.8.

Use of the risk factors from Section 2.1.2.9 (Table 2.1-3) with the doses calculated for this scenario results in the calculated health effects shown in the last three columns of Table 3.1.3.3-8. The likelihood that a facility worker would experience a health effect in the 50 years following the postulated accident is about one in 400. For a member of the public, the likelihood ranges from about one in 270 for the maximally exposed individual at the nearest site boundary to about one

Table 3.1.3.3-6. Total radiological source term for aircraft crash at Hot Fuel Examination Facility.

Nuclide	Source term (Ci)	Nuclide	Source term (Ci)
Krypton-85	2.70E+03	Xenon-133	1.38E+05
Strontium-89	3.93E+00	Cesium-134	3.63E+02
Strontium-90	6.41E+00	Cesium-136	1.45E+01
Yttrium-90	6.41E+00	Barium-137m	3.83E+00
Yttrium-91	1.99E+02	Cesium-137	2.51E+03
Zirconium-95	2.81E+02	Barium-140	1.36E+02
Niobium-95	3.54E+02	Lanthanum-140	1.63E+02
Molybdenum-99	7.90E+00	Cerium-141	2.44E+02
Technetium-99m	7.90E+00	Cerium-144	1.29E+02
Rubidium-103	4.48E+02	Praseodymium-143	1.37E+02
Rubidium-106	9.14E+01	Praseodymium-144	1.29E+02
Rhodium-103m	2.29E+02	Praseodymium-144m	5.73E-01
Rhodium-106	5.46E+01	Neodymium-147	4.68E+01
Silver-111	3.12E+00	Promethium-147	2.98E+01
Antimony-125	1.95E+00	Europium-155	2.57E+00
Technetium-125m	1.35E-01	Europium-156	2.66E+00
Tellurium-127m	2.29E+00	Plutonium-238	5.88E-02
Tellurium-127	5.63E+00	Plutonium-239	5.83E-01
Tellurium-132	1.02E+01	Plutonium-240	2.54E-01
Iodine-131	2.63E+04	Americium-241	1.42E-02
Iodine-132	5.25E+03	Plutonium-241	1.55E+01

in 800 at locations about 80 kilometers (50 miles) from the accident. By selecting the east-southeast compass sector as the worst case in the maximum sector population of 26,000 within an 80-kilometer radius of the accident, up to 2 health effects (including 1 fatal cancer) are calculated to occur over the lifetime of the exposed population.

To reduce the dose delivered to a hypothetical individual at the nearest site boundary below the protective action guide of 5 rem, it would be necessary to evacuate the individual and interdict the consumption of food grown at the site boundary within 180 days of the accident. Table 3.1.3.3-8 reflects this protective measure. No other members of the public would receive doses greater than the protective action guide.

Table 3.1.3.3-8. Summary of dose calculation results for aircraft crash at Hot Fuel Examination Facility at Argonne National Laboratory-West.

Receptor location	Inhalation CEDE[a] (rem)	Air immersion EDE[a] (rem)	Ground surface EDE (rem)	Ingestion CEDE (rem)	Total EDE (rem)	Likelihood of fatal cancer	Total likelihood of health effect
Facility worker (100 m)	4.5E+00	4.1E−02	5.7E-02	NA	4.6E+00	1.8E-03	2.6E-03
U.S. 20/26 (5,240 m)	3.2E−01	3.2E-03	2.1E-03	NA	3.2E−01	1.6E-04	2.3E−04
Nearest site boundary (5,240 m)	3.2E−01	3.2E-03	3.0E−01	4.4E+00[b]	5.0E+00	2.5E-03	3.7E−03
Dose to maximally exposed individual at nearby communities (rem)							
Atomic City (21 km)	1.1E−01	9.7E-04	1.1E−01	3.1E+00	3.3E+00	1.7E-03	2.4E-03
Mud Lake/Terreton (32 km)	8.5E−02	6.8E−04	8.0E−02	2.4E+00	2.6E+00	1.3E-03	1.9E-03
Howe (35 km)	8.0E−02	6.4E-04	7.5E−02	2.2E+00	2.4E+00	1.2E-03	1.8E-03
Idaho Falls/ Blackfoot (50 km)	6.5E-02	4.9E-04	6.1E-02	1.8E+00	1.9E+00	9.5E-04	1.4E-03
Arco (52 km)	6.4E-02	4.7E-04	6.0E-02	1.8E+00	1.9E+00	9.5E-04	1.4E-03
Rigby (60 km)	5.9E-02	4.3E-04	5.5E-02	1.6E+00	1.7E+00	8.5E-04	1.2E−03
Craters of the Moon National Monument (73 km)	5.3E-02	3.8E-04	5.0E-02	1.5E+00	1.6E+00	8.0E-04	1.2E-03
Rexburg (74 km)	5.3E-02	3.8E-04	5.0E-02	1.5E+00	1.6E+00	8.0E-04	1.2E-03
Dose to maximum sector population within 80-km (50-mile) radius (person-rem)						Number of fatal cancers	Total number of individuals experiencing health effects
Maximum sector population 26,000 95%	7.0E+02	5.1E+00	6.6E+02	6.2E+02	2.0E+03	1.0E+00	1.5E+00
Maximum sector population 5,100 50%	6.5E+01	5.5E-01	6.2E+01	5.8E+02	7.1E+02	3.6E-01	5.2E-01

a. EDE — effective dose equivalent; CEDE — committed EDE.

a. Protective actions are assumed to have taken place to limit dose to affected population to 5 rem in accordance with protective action guidelines.

management of waste fission products generated in the fuel cycle processes conducted in the Argon Cell.

At the date of this analysis, modification of the facility is under way to support the Integral Fast Reactor program. The accident analysis is based on the proposed new mission of the Fuel Cycle Facility. The fuel cycle process conducted in the Fuel Cycle Facility is briefly described below.

Program fuel cycle operations for the Integral Fast Reactor primarily involve two hot cells in the Fuel Cycle Facility: the Air Cell and the Argon Cell (Figures 3.1.3.4-1 and 3.1.3.4-2). The facility incorporates a contaminated equipment wash/repair area in the basement, in which equipment from the cells can be washed/decontaminated and repaired, either by suited entry or through a glove wall. Associated facilities include the cask transfer tunnels, a high bay, operating corridor, laboratories and offices, and support equipment. The Air Cell has an air atmosphere, and is used for receipt of fuel subassemblies from Experimental Breeder Reactor II (EBR-II), disassembly and assembly of fuel components, transfers of fuel into and out of the Argon Cell, and interim storage of up to 50 fuel subassemblies. No operations intended to penetrate fuel rod cladding are conducted in the Air Cell.

The main fuel cycle operations take place in the Argon Cell. Fuel is transferred to the Argon Cell from the Air Cell, chopped up in 10-kilogram (22-pound) batches, melted, electrorefined to separate fissile material from fission products, and cast into new fuel elements. The new fuel elements are transferred to the Air Cell for assembly into fuel subassemblies for reuse in the EBR-II reactor. Fission product waste material is in a solidified salt form, and stored in the Argon Cell in sealed storage cans below the floor level.

The major events in an aircraft crash accident scenario are as follows:

- A large aircraft crashes directly into the Fuel Cycle Facility (Building 765).
- The impact has sufficient force to cause catastrophic failure of the building structure and to breach the Argon Cell.
- The fuel in the aircraft is released to the facility and is ignited.
- The ensuing fire involves the exposed contents of the Argon Cell, resulting in atmospheric release of radionuclides.

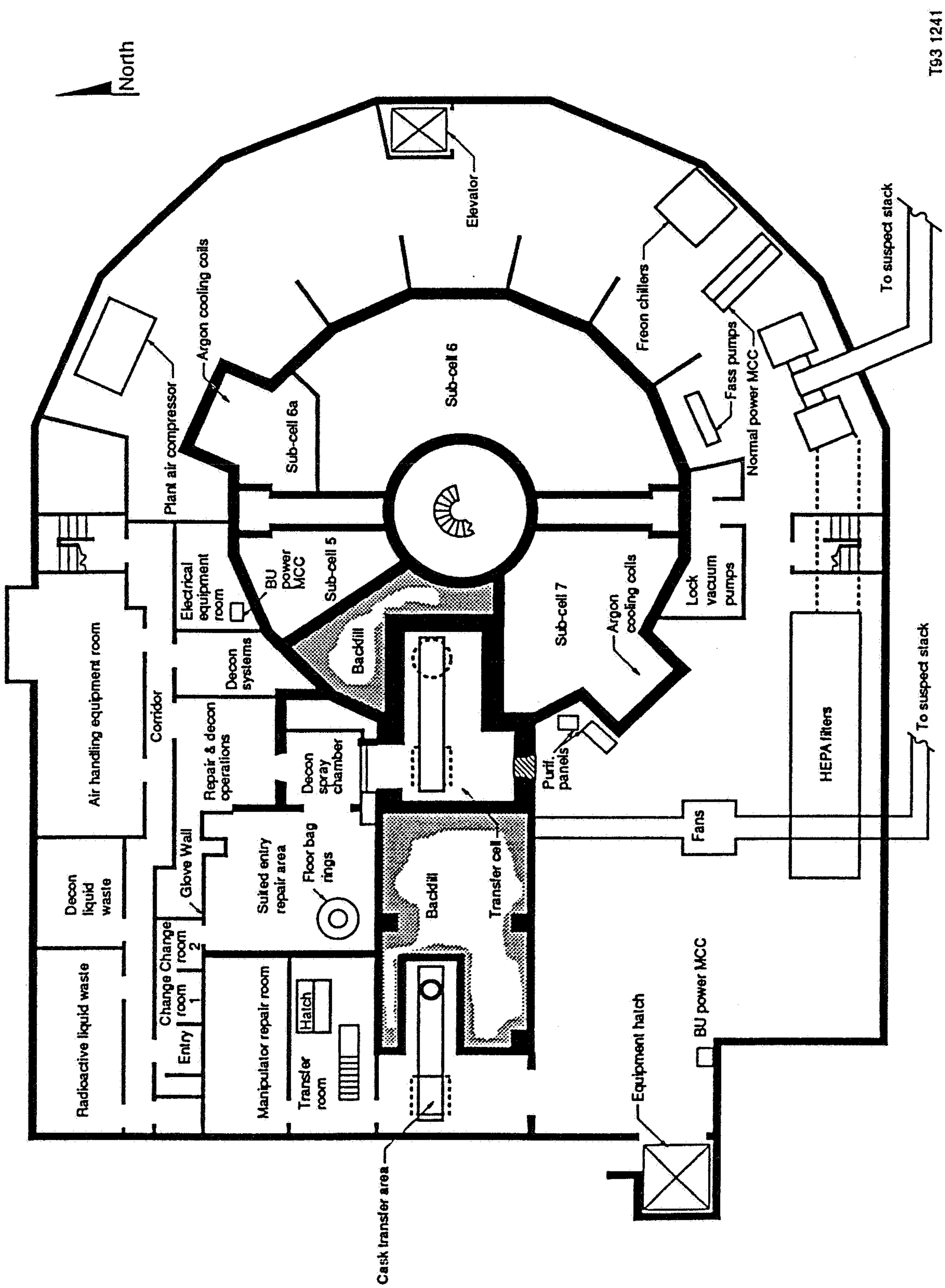

Figure 3.1.3.4-2. Fuel Cycle Facility basement floor plan.

Table 3.1.3.4-1. Radiological material at risk for aircraft crash at Fuel Cycle Facility.

Material	Damage ratio	Involved material	Curies
Exposed fuel (45 kg)[a]	1.0	45 kg (equivalent of 12.2 subassemblies)	Plutonium: Table 3.1.3.4-2 Fission products: Table 3.1.3.4-3
Fission product salt waste canisters[b]	<0.25	1 canister	Table 3.1.3.4-4
Krypton-85[c]	1.0	1,810 Ci	1,810[c]

To convert kilograms to pounds, multiply by 2.2.

a. 90-day decay since reactor shutdown.

b. Argon Cell, below floor level storage, 10 year cooling assumed.

c. Accumulated in Main Cell atmosphere (ANL-W 1993).

Table 3.1.3.4-2. Bounding transuranic elements isotopic mass in one subassembly in Argon Cell in Fuel Cycle Facility.[a]

Plutonium isotope	Mass (g)	Curies
Plutonium-238	4.40E-01	7.57E+00
Plutonium-239	7.45E+02	4.59E+01
Plutonium-240	2.26E+02	5.14E+01
Plutonium-241	2.00E+01	2.03E+03
Plutonium-242	6.60E+00	2.54E-02
Americium-241	2.50E+00	8.65E+00

To convert grams to pounds, multiply by 2.2×10^{-3}.

a. Data from ANL-W (1993). **Note:** As stated in that reference, these values are triple the transuranic inventory normally expected in one subassembly to provide a margin for potential future Integral Fast Reactor process modifications.

Table 3.1.3.4-4. Radionuclide inventory in one salt waste canister--10-year decay.[a]

Nuclide	Activity in one salt waste can (Ci)
Strontium-90	5.95E+04
Yttrium-90	5.95E+04
Cesium-134	9.19E+02
Barium-137m	8.92E+04
Cesium-137	8.92E+04
Plutonium-238	5.64E+00
Plutonium-239	3.53E+01
Plutonium-240	4.01E+01
Plutonium-241	1.03E+03
Plutonium-242	2.00E-02
Americium-241	2.58E+01

a. Data from ANL-W (1993).

The source term is developed by multiplying the appropriate quantities summarized in Table 3.1.3.4-1 by the airborne release fractions, respirable fractions, and leak path factors summarized in Table 3.1.3.4-5. Total resulting source terms for the aircraft crash accident are specified in Table 3.1.3.4-6. The values reported in the tables reflect the material that is involved in the accident; i.e., 45 kilograms (100 pounds) of 90-day cooled fuel, one fission product salt waste canister, and krypton-85 in the Argon Cell atmosphere at the time of the accident.

3.1.3.4.3 Dose Calculations and Results—The RSAC-5 (Wenzel 1993) model referenced here is described in more detail in Section 2.1.2. That section also documents the generic assumptions and parameters used in the accident assessments. By referencing the information in Section 2.1.2, the discussion below is limited to the unique aspects of the model and specific results for this bounding accident--radiological release due to a large aircraft crash into ANL-W Fuel Cycle Facility.

Table 3.1.3.4-6. Total radiological source term for aircraft crash at Fuel Cycle Facility.

Nuclide	Source term (Ci)	Nuclide	Source term (Ci)
Krypton-85	1.81E+03	Tellurium-129m	5.60E-01
Strontium-89	1.40E+01	Cesium-134	1.16E+03
Strontium-90	6.00E+02	Barium-137m	9.00E+02
Yttrium-90	6.00E+02	Cesium-137	6.14E+03
Yttrium-91	2.15E+01	Barium-140	2.47E-01
Zirconium-93	2.06E-04	Lanthanum-140	2.88E-01
Zirconium-95	3.38E+01	Cerium-141	9.88E+00
Niobium-95	5.85E+01	Cerium-144	6.10E+01
Niobium-95m	4.30E-01	Praseodymium-143	3.55E-01
Technetium-99	9.88E-04	Praseodymium-144	6.10E+01
Ruthenium-103	1.32E+01	Neodymium-147	3.73E-02
Ruthenium-106	3.05E+01	Promethium-147	1.98E+01
Rhodium-103m	1.32E+01	Promethium-148	2.65E-02
Silver-110m	2.06E-02	Promethium-148m	3.78E-01
Silver-111	1.07E-04	Samarium-151	2.31E-01
Cadmium-115m	2.39E-02	Europium-154	9.88E-02
Tin-119m	1.07E-02	Europium-155	7.85E-01
Tin-123	3.30E-01	Europium-156	1.07E-02
Tin-125	2.06E-04	Plutonium-238	2.45E-02
Antimony-125	1.07E+00	Plutonium-239	1.49E-01
Antimony-126	2.15E-04	Plutonium-240	1.67E-01
Tellurium-125m	2.65E-01	Americium-241	3.30E-02
Tellurium-127m	6.20E-01	Plutonium-241	6.45E+00
Tellurium-127	6.00E-01	Plutonium-242	7.80E-05
Tellurium-129	3.55E-01		

For this scenario, the RSAC-5 program was used to determine the dose from external and internal pathways at two receptor locations: (1) a location within the INEL controlled access zones (inside the ANL-W complex) and (2) the nearest public access and nearest site boundary at U.S. Highway 20, generally south of ANL-W. At the nearest public access/nearest site boundary location, doses are calculated both for the stranded motorist and for the individual living at the nearest site boundary. The assumptions for the receptors are the same as most other accidents and as summarized in Section 2.1.2.7. Nearest site boundary assumptions and parameters were used in RSAC-5 to calculate individual doses beyond the nearest site boundary.

Table 3.1.3.4-8. Summary of dose calculation results for aircraft crash at Fuel Cycle Facility at Argonne National Laboratory-West.

Receptor location	Inhalation CEDE[a] (rem)	Air immersion EDE[a] (rem)	Ground surface EDE (rem)	Ingestion CEDE (rem)	Total EDE (rem)	Likelihood of fatal cancer	Total likelihood of health effect
Facility worker (100 m)	3.6E+00	6.7E−03	1.5E−03	NA	3.6E+00	1.4E−03	2.0E−03
U.S. 20/26 (5,240 m)	2.5E−01	6.7E−04	6.7E−05	NA	2.5E−01	1.3E−04	1.8E−04
Nearest site boundary (5,240 km)	2.5E−01	6.7E−04	2.7E−01	1.3E+00	1.8E+00	9.0E−04	1.3E−03
Dose to maximally exposed individual at nearby communities (rem)							
Atomic City (21 km)	8.8E−02	2.4E−04	9.7E−02	4.5E−01	6.4E−01	3.2E−04	4.7E−04
Mud Lake/Terreton (32 km)	6.7E−02	1.8E−04	7.4E−02	3.5E−01	4.9E−01	2.5E−04	3.6E−04
Howe (35 km)	6.4E−02	1.7E−04	7.0E−02	3.3E−01	4.6E−01	2.3E−04	3.4E−04
Blackfoot/Idaho Falls (50 km)	5.2E−02	1.4E−04	5.7E−02	2.7E−01	3.8E−01	1.9E−04	2.8E−04
Arco (52 km)	5.1E−02	1.4E−04	5.6E−02	2.6E−01	3.7E−01	1.9E−04	2.7E−04
Rigby (60 km)	4.7E−02	1.3E−04	5.2E−02	2.4E−01	3.4E−01	1.7E−04	2.5E−04
Craters of the Moon National Monument (73 km)	4.3E−02	1.2E−04	4.7E−02	2.2E−01	3.1E−01	1.6E−04	2.3E−04
Rexburg (74 km)	4.3E−02	1.2E−04	4.7E−02	2.2E−01	3.1E−01	1.6E−04	2.3E−04
Dose to maximum sector population within 80-km (50-mile) radius (person-rem)						Number of fatal cancers	Total number of individuals experiencing health effects
Maximum sector population 26,000 95%	5.3E+02	1.4E+00	6.1E+02	1.5E+03	2.6E+03	1.3E+00	1.9E+00
Maximum sector population 5,100 50%	5.1E+01	1.3E-01	5.5E+01	1.4E+02	2.5E+02	1.3E-01	1.8E-01

a. EDE — effective dose equivalent; CEDE — committed EDE.

Alternative C would involve shipping the majority of INEL spent nuclear fuel offsite. basis of the increased handling, characterization, and packaging necessary accomplish offsite shipment, it was assumed that the frequency, but not the consequences, of a fuel handling acc would increase by a factor of approximately 8.6 over Alternative A.

Under Alternative D, storage of spent nuclear fuel from the DOE complex would be centralized at the INEL. The handling of spent nuclear fuel at INEL would increase by a fact approximately 20 over that in Alternative A. On the basis of these factors, it was assumed th frequency, but not the consequences of a fuel handling or inadvertent criticality accident woul increase ten-fold over Alternative A. Accidents associated with spent fuel processing were ev for Alternative D (Table 3.2-2). The criticality accident was assumed to involve 3×10^{19} fi (more severe than the 1×10^{19} fission accident evaluated for Alternatives A, B, and C). Th processing accidents evaluated were a dissolver hydrogen explosion and inadvertent dissolutio day cooled fuel. The scaling factors used for spent nuclear fuel accidents are summarized in 3.2-1. The accidents that differ from those analyzed under Alternatives A, B, and C are sum in Table 3.2-2.

Table 3.2-1. Spent nuclear fuel accidents scaling factors—Alternatives B, C, and D.[a]

Accident[b]	Alternative	Frequency scaling factor	Consequence scalin
Fuel handling accident, fuel pin breach, venting of noble gasses and iodine (bounding fuel handling accident)	B	4.8	1
	C	8.6	1
	D	20	1
Uncontrolled chain reaction (criticality) accident at Idaho Chemical Processing Plant	B	1	1
	C	1	1
	D	1	3×10^{19} vs. 1 › fissions
Severe seismic event, cell breach, and fuel melting at Hot Fuel Examination Facility (HFEF) at Argonne National Laboratory-West (ANL-W)	B	1	1
	C	1	1
	D	1	1
Aircraft crash into HFEF at ANL-W	B	1	1
	C	1	1
	D	1	1

a. Multiply the frequency and consequence of the accident analyzed under Alternative A by the inc scaling factor to obtain the risk of the accident under each alternative.
b. These accidents were found bounding in the abnormal, design basis, and beyond design basis fre categories and, as such, characterize risks from management of spent nuclear fuel.

The Atmospheric Protection System (in CPP-649 and CPP-756) and the ICPP Main Stack (CPP-708) make up a final treatment and release point for several process and ventilation systems (see Figure 4.1.2.1-1). The system has two trains: the ventilation train handles only ventilation air; the process train handles vessel offgas (from possible processing of in-process inventory) and calciner process offgas. (Both trains are designed to handle some dissolver offgases, but no further dissolver operations are planned.)

Standard postulated accidents are identified in the literature (Brynda et al. 1986) to be considered for offgas treatment systems of reprocessing plants and of liquid waste treatment facilities, such as calciners. These accidents include failure of filter, loss of differential pressure control in ventilation zones, and failure of iodine scrubber system (applicable only for dissolver operations).

The potential material at risk in the Atmospheric Protection System is primarily the radionuclide inventory entrained in the following filter components:

- CPP-756 ventilation prefilter
- 104 ventilation high-efficiency particulate air (HEPA) filters in CPP-649
- Nine Atmospheric Protection System process HEPA filters in CPP-649.

The accident is limited to the radionuclide inventory in both the ventilation prefilter and the 104 ventilation offgas HEPA filters in the Atmospheric Protection System for the following reasons: (a) the process train of the system has no connections with the ventilation train before the common discharge at the stack, (b) the two trains are separated by a reinforced concrete wall in CPP-649, (c) the projected inventory in the ventilation train of the system bounds that in the process train, and (d) the ventilation prefilter and HEPA filters are in relatively close proximity, with no dampers in the ducting separating them. (Eliminating the inventory of the process HEPA filters from the material at risk is a qualitative application of the damage ratio defined in Section 2.1.2.4.)

The postulated mechanism for uncontrolled release of the filter inventory is a sustained or intense fire in the filter media. Other mechanisms such as explosions or filter failures could cause a release from the filters, but the release due to these mechanisms is expected to be bounded by the release due to a fire.

The ventilation offgas flows through the following Atmospheric Protection System components (see Figure 4.1.2.1-1):

- Prefilter in vault CPP-756

- HEPA Filter Caissons (26) in CPP-649
- Blowers WL-210, -211, -212 in CPP-605
- Exhaust Stack CPP-708, entering at (5 meters) 16 feet above the base.

The CPP-756 prefilter is located in an underground reinforced concrete vault adjacent to CPP-649, which is divided into four parallel bays. The prefilter media consists of five layers of varying density, separately supported, packed fiberglass. The prefilter is estimated to have an efficiency of 90 percent for removal of particles greater than 0.7 micron based on operating experience of a similar filter at the Hanford site (Robson 1992a).

The HEPA filters in the Atmospheric Protection System are fabricated of fiberglass with stainless-steel frames, and are rated for temperatures up to 370°C (700°F). Four parallel filters are installed in each of 26 parallel caissons made of stainless steel. The HEPA filters have rated efficiency of 99.97 percent for particles greater than or equal to 0.3 micron in diameter as tested with dioctyl phthalate (DOP) smoke.

The major events in this accident scenario are as follows:

- A process upset or other accident upstream of the Atmospheric Protection System results in release of flammable gas, vapors, or dust to the ventilation system.
- The flammable mixture is ignited in or near the filters by static electricity or other means.
- The forced ventilation flow and/or convective stack "draw" cause propagation of the fire downstream through the ventilation prefilter and 104 HEPA filters.
- The fire causes resuspension or volatilization of a fraction of the radionuclide inventory in the Atmospheric Protection System ventilation filter train, releasing the inventory uncontrolled to the atmosphere through the stack.

An upper bound for the frequency of releasing a combustible mixture to the Atmospheric Protection System is 8.3×10^{-3} per year based on 40 facility-years with no release. The likelihood of static discharge is considered to be 1.0 with the presence of the combustible mixture. The probability of release of 0.01 percent of the inventory in particulate or volatilized form (as assumed below in development of the source term) is less than 1×10^{-2}, as estimated by engineering judgment applied to HEPA filter test data (Ammerich et al. 1988). Therefore, the ICPP Atmospheric Protection System release accident is estimated to be in likelihood range of 1×10^{-4} to 1×10^{-6} per year.

(e) The source term is developed by multiplying the material at risk by the airborne release fraction, leak path factor, and respirable fraction (see Section 2.1.2.4). (This material at risk includes the damage ratio, as discussed in Section 4.1.2.1.1) Since these multipliers, except the airborne release fraction, are 1.0, the source term is equal to 1 percent of the material at risk.

Total source term for the fire, as calculated by the process involving the Microshield code, is shown in Table 4.1.2.1-1. The source term consists of the following radionuclides: strontium-90, yttrium-90, ruthenium-106 and its short-lived daughter rhodium-106, antimony-125, cesium-134, cesium-137 and its short-lived daughter barium-137m, uranium-234, plutonium-239, and americium-241.

4.1.2.1.3 Dose Calculations and Results—The RSAC-5 model referenced here (Wenzel 1993) is described in more detail in Section 2.1.2.1. That section also documents generic assumptions and parameters used in the accident assessments. By referencing the information in Section 2.1.2, the discussion below is limited to the unique aspects of the model and specific results for this bounding accident--fire in the ICPP Atmospheric Protection System. For this scenario, the RSAC-5 program was used to determine the dose from external and internal pathways at three receptor locations. The receptor locations are (1) a location within the INEL controlled access zones (inside the ICPP facility area), (2) the nearest public access at U.S. Highway 20/26, and (3) the nearest site boundary generally south of ICPP. The assumptions for these receptors are the same as most other accidents and as summarized in Section 2.1.2.7. Nearest site boundary assumptions and parameters were used in RSAC-5 to calculate individual doses beyond the nearest site boundary.

The release duration is 60 minutes. The meteorological/dispersion parameters for RSAC-5 are as described in Section 2.1.2.5 and summarized in Table 2.1-20, except the release is not at ground level. The Atmospheric Protection System provides an elevated discharge of the source term from the main stack. The biological parameters for RSAC-5 are as described in Section 2.1.2.6 and exposure times do not deviate from Table 2.1-21. Scenario-specific input parameters for RSAC-5 are given in Table 4.1.2.1-2.

The dose results for each applicable pathway and for their total at each of the three receptor locations are given in Table 4.1.2.1-3. Dose results are also presented for various communities within 80 kilometers (50 miles) of the ICPP. The table also includes the dose for the population within an 80-kilometer radius, as calculated by the methodology described in Section 2.1.2.8.

Use of the risk factors from Section 2.1.2.9 (Table 2.1-3) with the doses calculated for this scenario results in the calculated health effects shown in the last two columns of Table 4.1.2.1-3.

Table 4.1.2.1-2. Meteorological/dispersion parameters used in dosimetry calculations for filter bank fire in Atmospheric Protection System.

Parameter	Facility worker	Nearest public access (U.S. 20/26)	Nearest site boundary[b]
Receptor distance (m)	100	5,870[a]	14,000
Release elevation (m)	76.8	76.8	76.8
Release duration (min.)	60	60	60

To convert from meters to feet, multiply by 3.28.

a. Source: Section 2.1.2.7 (Table 2.1-2).

b. Nearest site boundary values (except receptor distance) also used in calculations of doses beyond nearest site boundary.

Given the accident occurs, the likelihood that a facility worker would experience a health effect in the 50 years following the postulated accident is about one in 10,000. For a member of the public, the likelihood ranges from less than one in 100,000 at the nearest public access down to about one in a million for a location about 80 kilometers (50 miles) from the accident. In the population of 24,000 in the most populous sector within an 80-kilometer radius of the accident, less than one health effect is calculated to occur in the 50 years following the accident.

4.1.2.1.4 Preventive and Mitigative Measures—The following features of the Atmospheric Protection System are designed to prevent a fire in the HEPA filters:

- Filters are fabricated of fire-resistant materials with metal frames, HEPA filters rated for temperatures up to 370°C (700°F).
- Filter caissons, ducting, and supports are constructed of metal.
- There are no ignition sources and combustible material in the ventilation offgas flow path.
- High ventilation flow rate is designed to prevent occurrence of a flammable mixture of gases or vapors.
- The prefilter vault (CPP-756), and the Atmospheric Protection System Building (CPP-649), is constructed of reinforced concrete and steel.

The following measures are designed to mitigate the consequences of a fire in the HEPA filters:

- Local fire protection equipment is installed in CPP-649 (fire hose racks, fire extinguishers).
- Radiation monitoring equipment and alarms are installed in the Main Stack offgas flow train, and elsewhere throughout ICPP.
- Emergency response facilities and procedures are in place for ICPP, INEL, and surrounding area.
- DOE Fire Department at Central Facilities Area is available on short notice.

Credit is taken for the radiation alarms and the emergency response procedures by assuming the ICPP workers are exposed to the airborne plume for only 5 minutes and to ground surface contamination for only 20 minutes.

4.1.2.2 Idaho Chemical Processing Plant: Earthquake-Induced Main Stack Collapse. The potential consequences of the toppling of the Main Stack at the Idaho Chemical Processing Plant (ICPP) caused by a large seismic event are analyzed.

4.1.2.2.1 Description of Accident—The bounding airborne radiological release due to this toppling event results from crushing of filter equipment containing radionuclides, and is described in this section. Other types of releases could occur from the stack toppling; however, postulated releases are bounded by the releases considered here.

The ICPP Main Stack is the final offgas release point for gaseous waste streams from several ICPP facilities. The offgas is passed through treatment equipment--mist eliminators, condensers, prefilters, and high-efficiency particulate air (HEPA) filters--before entering the stack. The stack is 76 meters (250 feet) high with an 2.4-meter (8-feet) inside diameter tapering to 2 meters (6.5 feet) at the top. It is constructed of a stainless-steel liner encased in a layer of foam, a layer of brick, and two external sheaths of reinforced concrete. The total weight of the stack is over 454,000 kilograms (4 million pounds).

If the stack is assumed to topple in any direction and if the leverage of the stack base is included, the collapsed stack could impact the area within a 91-meter (300-foot) radius circle with the center at the stack base (Chung 1993). The primary facilities of concern within this 300-foot radius circle are shown in Figure 4.1.2.2-1. The analysis determined (a) the potential inventory of hazardous and radioactive material in each facility, and (b) the potential damage that could occur to the facilities in a stack toppling accident.

The damage analysis (Chung 1993) indicates that major damage would generally result from stack impact. This analysis identified the various types of bounding scenarios for a stack toppling event:

- Bounding radiological airborne release:
    - CPP-604 vessel offgas HEPA filter F-WL-121 and CPP-756 Atmospheric Protection System ventilation prefilter crushing accident—west of the stack (see Figure 4.1.2.2-1)
- Nonradiological airborne release:
    - Not analyzed further; bounded by the chlorine release considered in Section 7.1.3.1.
- Radiological and nonradiological release to groundwater:
    - Not analyzed further; bounded by the high-level waste tank failure accident (see Sections 4.1.2.4 and 7.1.2.5).

The postulated events leading to the bounding atmospheric release of radionuclides are as follows:

- The maximum earthquake event results in a peak ground acceleration sufficient to cause toppling of the Main Stack.
- The Main Stack falls to the west, crushing the CPP-604 vessel offgas filter and CPP-756 ventilation prefilter.[a]

The mechanism causing atmospheric release of particulate material is failure of the filter confinement barrier and physical impact of the stack on the prefilter.

One can conservatively assert that the stack would be more likely to fall in a direction parallel to the direction of travel of incoming seismic waves, and perpendicular to the waves themselves. Seismic waves approaching the ICPP from the Lemhi fault north and west of the ICPP may make it more likely for the stack to fall in the west northwest direction. The frequency of occurrence of a seismic event large enough to cause failure of the Main Stack is estimated at 3×10^{-4}.

a. Since the stack could theoretically fall in any direction, selecting the direction is a qualitative application of the damage ratio described in Section 2.1.2.4.

In this scenario the outer containment is assumed breached, allowing the unmitigated release of 100 percent of the resuspended constituents to the atmosphere. The leak path factor (defined in Section 2.1.2.4) for the scenario is therefore 1.0.

The source term for this accident scenario is developed by multiplying the material at risk as follows:

Source term = material at risk × airborne risk factor × leak path factor × respirable fraction.

Because the airborne respirable fraction is 0.1 and the leak path factor and respirable fraction are both 1.0, the source term is equal to 0.01 times the material at risk shown in Table 4.1.2.2-1. The source term for major constituents in the ICPP Main Stack toppling scenario is shown in Table 4.1.2.2-2.

The primary conservatisms included in the source term for the stack toppling bounding accident analysis are as follows:

- The assumed particulate loading on the filters is at the reasonably foreseeable maximum through the year 2005.

- One percent of the particulate loading on the filters is assumed to become airborne in the accident (DOE 1993c). Because the vessel offgas HEPA filter is located in a cell belowground and the Atmospheric Protection System prefilter is in a concrete underground vault covered by 2 feet (0.7 meter) of soil, this release fraction is considered conservative.

Table 4.1.2.2-2. Important contributors to source term for Main Stack toppling accident at Idaho Chemical Processing Plant.

Nuclide	Source term (Ci)
Strontium-90	0.0644
Yttrium-90	0.0644
Ruthenium-106	0.226
Rhodium-106	0.226
Antimony-125	0.366
Cesium-134	0.0152
Barium-137M	0.136
Cesium-137	0.147
Uranium-234	0.2540
Plutonium-239	0.022
Americium-241	0.0015

The dose results for each applicable pathway and for their total at each of the three receptor locations are given in Table 4.1.2.2-4. Dose results are also presented for various communities within 80 kilometers (50 miles) of the ICPP. The table also includes the dose for the population with a 50-mile radius, as calculated by the methodology described in Section 2.1.2.8. Because of the conservatively short 15-minute exponential release from the stack impact, a maximally exposed worker at 100 meters (328 feet) is exposed to 89 percent of the plume release in five minutes.

Structures in the line of the stack would receive major damage. Collapse of the stack would necessitate immediate shutdown of all operating facilities served by the stack. Before resumption of normal operation at any ICPP facilities, extensive ground surface decontamination of the ICPP site area may be necessary due to deposition of particulate from the accident.

Use of the risk factors from Section 2.1.2.9 (Table 2.1-3) with the doses calculated for this accident results in the calculated health effects shown in the last two columns of Table 4.1.2.2-4.

4.1.2.2.4 Preventive and Mitigative Measures—The Main Stack has received various structural upgrades during its operating history to prevent its collapse or toppling during an earthquake. However, analysis has shown that the stack will fail during a large seismic event (Chung 1993). Impact of the 1.7-million-kilogram (4-million pound) stack is beyond the design basis of most of the surrounding facilities, and such an event would be expected to result in major damage.

4.1.2.3 Idaho Chemical Processing Plant: Earthquake-Induced Structural Failure of Calcined Solids Storage Facility. The response of the Calcined Solids Storage Facility at the Idaho Chemical Processing Plant (ICPP) to a large earthquake was investigated. This facility is described in Section 4.1.3.1, where the radiological consequences of an aircraft crash into the facility are discussed.

4.1.2.3.1 Description of Accident—The earthquake used for the accident analysis has a return period of 100,000 years. Previous analyses given in the ICPP Plant Safety Document (WINCO 1992b) suggest that no gross structural damage is expected to the bins and vaults for earthquakes up to about this magnitude. However, undetected, defective welds have been cited as a possible failure mechanism during earthquakes.

The postulated events leading to atmospheric release of radionuclides and toxic chemicals are as follows:

- The maximum earthquake results in a peak horizontal acceleration beyond the design basis of the facility.

- A severely defective bin weld in one Calcined Solids Storage Facility bin fails from stress induced by the earthquake, releasing the calcine from that bin to the vault.

- This release causes a portion of this calcine to become airborne in the vault.

- The vault cooling air system is disabled due to loss of power or structural damage such that the HEPA filters cannot be aligned to the cooling air system. (The cooling system normally exhausts unfiltered air to the environment and switches to a filtered release path only upon detection of high radioactivity in the air.)

- Although the vault remains intact after the earthquake, airborne calcine is released by natural convection through the vault cooling air system.

The frequency of occurrence for the seismic event is 1×10^{-5} per year. The conditional probability of a release given the earthquake is conservatively assumed to be one.

4.1.2.3.2 Development of Source Term—The following assumptions were made to develop the source term:

(a) The earthquake has sufficient force to cause failure of a defective weld of one bin in the Fifth Calcined Solids Storage Facility. The radionuclide inventory (curies) in the entire contents of all seven bins (1.46 million kilograms--330 tons) is the material at risk. The damage ratio is one-seventh.

Concentrations in the material at risk during the earthquake are summarized in Tables 4.1.3.1-1 (radiological) and 7.1.3.1-1 (nonradiological).

The next three assumptions result in an airborne release fraction of 4×10^{-5}:

(b) 100 percent of the calcine contents in the bin is released to the vault.

(c) As for the aircraft crash accident, 40 percent of the calcine released to the vault is fines (Berreth 1988).

(d) 0.01 percent of these calcine fines becomes airborne as a result of calcine settling after the accident. This amount is about equivalent to the amount that could be suspended in the free vault volume at maximum values determined experimentally by Schwendiman (1977). No subsequent driving force exists to suspend additional material.

Table 4.1.2.3-2. Total release fractions for earthquake at the Calcined Solids Storage Facility.

Material	Airborne release fraction	Respirable fraction	Leak path factor	Total
Solid calcine	4.0E-05[a]	5.0E-02[b]	1.0[c]	2.0E-06[d]

a. Based on 40 percent fines (Berreth 1988) and 0.01 percent of fines becoming airborne (Schwendiman 1977).

b. Source: Berreth (1988). Applied only to the source term for the inhalation pathway.

c. A leak path factor of 1.0 is assigned for a major failure of confinement barriers.

d. Airborne release fraction × respirable fraction × leak path factor.

Table 4.1.2.3-3. Total radiological source term for earthquake at the Calcined Solids Storage Facility.

	Source term (Ci)			Source term (Ci)	
Nuclide	Total	Respirable	Nuclide	Total	Respirable
Strontium-90	2.2E+01	1.1E+00	Neptunium-237	5.6E-05	2.8E-06
Yttrium-90	2.2E+01	1.1E+00	Plutonium-238	1.34E-01	6.7E-03
Ruthenium-106	5.4E-03	2.7E-04	Plutonium-239	2.8E-03	1.4E-04
Cesium-134	1.84E-01	9.2E-03	Plutonium-240	1.74E-03	8.7E-05
Cesium-137	2.4E+01	1.2E+00	Plutonium-241	4.0E-01	2.0E-02
Barium-137m	2.4E+01	1.2E+00	Plutonium-242	2.4E-06	1.2E-07
Promethium-147	3.8E+00	1.9E-01	Americium-241	1.24E-03	6.2E-05
Europium-154	1.84E-01	9.2E-03	Americium-243	6.0E-06	3.0E-07

Table 4.1.2.3-4. Nonradiological source term for earthquake at the Calcined Solids Storage Facility.

Constituent	Source term (mg)
ZrO_2	8.28E+04
CdO	4.16E+03
Hg	4.16E+03
Cr	4.16E+03

Table 4.1.2.3-6. Summary of dose calculation results for earthquake at the Calcined Solids Storage Facility at Idaho Chemical Processing Plant.

Receptor location	Inhalation CEDE[a] (rem)	Air immersion EDE[a] (rem)	Ground surface EDE (rem)	Ingestion CEDE (rem)	Total EDE (rem)	Likelihood of fatal cancer	Total likelihood of health effect
Facility worker (100 m)	1.2E+00	1.8E-03	3.5E-04	NA	1.2E+00	4.8E-04	6.7E-04
U.S. 20/26 (5,870 m)	2.3E-02	4.4E-05	4.4E-06	NA	2.3E-02	1.2E-05	1.7E-05
Nearest site boundary (14 km)	9.2E-03	1.7E-05	7.5E-03	5.9E-02	7.6E-02	3.8E-05	5.5E-05
Dose to maximally exposed individual at nearby communities (rem)							
Atomic City (17 km)	7.6E-03	1.4E-05	6.2E-03	4.8E-02	6.2E-02	3.1E-05	4.5E-05
Howe (24 km)	5.4E-03	1.0E-05	4.4E-03	3.4E-02	4.4E-02	2.2E-05	3.2E-05
Arco (31 km)	4.3E-03	8.1E-06	3.5E-03	2.7E-02	3.5E-02	1.8E-05	2.6E-05
Mud Lake/Terreton (49 km)	2.9E-03	5.4E-06	2.3E-03	1.8E-02	2.3E-02	1.2E-05	1.7E-05
Craters of the Moon National Monument (51 km)	2.8E-03	5.2E-06	2.3E-03	1.8E-02	2.3E-02	1.2E-05	1.7E-05
Blackfoot (62 km)	2.3E-03	4.4E-06	1.9E-03	1.5E-02	1.9E-02	9.5E-06	1.4E-05
Idaho Falls (72 km)	2.1E-03	3.9E-06	1.7E-03	1.3E-02	1.7E-02	8.5E-06	1.2E-05
Rigby (83 km)	1.9E-03	3.5E-06	1.5E-03	1.2E-02	1.5E-02	7.5E-06	1.1E-05
Dose to maximum sector population within 80-km (50-mile) radius (person-rem)						Number of fatal cancers	Total number of individuals experiencing health effects
Maximum sector population 24,000 95%	3.6E+02	3.4E-02	1.5E+01	5.6E+01	4.3E+02	2.2E-01	3.1E-01
Maximum sector population 9,100 50%	4.9E+00	4.9E−04	2.1E-01	7.8E-01	5.9E+00	3.0E−03	4.3E−03

a. EDE — effective dose equivalent; CEDE — committed EDE.

Four additional 114,000-liter (30,000-gallon) underground tanks rest on concrete pads with curbing (WM-103 through WM-106). These tanks are kept empty and may be used for waste storage only with prior DOE-Idaho Operations Office approval.

Buried waste lines carry high-level waste to the tanks for storage. All lines are fabricated of stainless steel and are enclosed in pipe encasements. Three types of encasements are used: split tile, stainless-steel-lined concrete, and stainless-steel pipe. Encasements drain to sumps instrumented to detect radioactivity. Routing of liquid waste within the tank farm area and from process areas to the tank farm is controlled by manually operated valves.

Based in part on the ICPP Plant Safety Document (WINCO 1992b), the following internal and external initiating events resulting in liquid or gaseous release were considered as potential candidates for the bounding accident:

- Tank leak due to corrosion, liquid release to vault
- Transfer line leak due to corrosion, liquid release to encasement
- Inadvertent tank overfill, liquid release to vault
- Hydrogen explosion, overpressure of offgas system
- Tank nuclear chain reactions, fission product release to offgas system
- Tank and vault failure due to an earthquake, liquid release to soil
- Transfer line and encasement failure due to an earthquake, liquid release to soil.

The tank/vault failure due to an earthquake was selected and other accidents eliminated for the reasons described in the following paragraphs.

The seismically initiated tank or vault failure is the bounding event among the liquid release events listed above. Events such as corrosion of tanks or transfer lines, or overfill of tanks would occur slowly, and would require extended operator error or instrument failure to cause significant liquid release outside the vault.

A tank explosion due to hydrogen would require not only buildup of an explosive hydrogen concentration in a tank, but also an ignition source. The chemistry of high-level waste at ICPP prevents formation of precipitates or crusts that have resulted in rapid release of hydrogen at other high-level waste storage tank facilities administered by DOE. The consequences of a hydrogen explosion are considered to be bounded by a tank failure.

On the basis of a review of previous structural analyses (Dearien 1971, Rahl 1976, EQE 1988, URS 1990, Advanced Engineering 1991), the following conclusions were reached regarding the structural integrity of the high-level waste tanks during the maximum earthquake (Chung 1992a):

- The steel 300,000-gallon tanks themselves can withstand the maximum earthquake. Some horizontal shifting of the tanks and shearing of pipe and instrument lines at the top of the tanks could occur.

- Vaults WM-180, WM-181, and WM-187 through WM-190, which were cast-in-place, may suffer small to moderate damage during the maximum seismic event, but collapse is not expected.

- Vaults WM-182 through WM-186 are assembled of precast concrete components. The method of construction of vaults WM-185 and WM-186 makes them more rugged than vaults WM-182 through WM-184. Vaults WM-182 through WM-184 could collapse upon the enclosed tank.

- The 18,400-gallon tanks WM-100, WM-101, WM-102, and WL-101 may be damaged and leak during the maximum earthquake. The concrete vault in which they are contained would not be damaged.

The major events in this accident scenario are:

- The maximum earthquake results in ground acceleration beyond the design basis of the precast concrete vault tanks.

- The seismic ground acceleration causes failure of one or more of tank vaults WM-182 through WM-184.

- Vault failure results in the failure of one (or the equivalent of one) 300,000 gal tank caused by inward falling pieces of the concrete vault impacting with the steel tank.

- Vault failure or the seismic event disables the vault sump pump.

- The entire contents of one (or the equivalent of one) tank is released unmitigated to the soil.

The annual frequency of occurrence of the maximum earthquake is on the order of 10^{-5}. For that earthquake, the likelihood of vault failure that results in drainage of the equivalent of one entire tank is conservatively assumed to be one.

personnel would be approximately 2 millirem, and doses to members of the public at U.S. Highway 20 and the nearest site boundary would be less than 1 millirem.

- Direct radiation doses to facility personnel are neglected due to the underground location of the tanks. Health physics personnel would be expected to become aware of the accident almost immediately, and any access to the area would be strictly controlled.

- The saturated hydraulic conductivity of the basalt supporting the vault is assumed to control the release rate of tank contents from the vault, resulting in a release time of 7.4 days.

- In support of the proposed High-Level Waste Tank accident analysis, an analysis was performed of postulated accidents in the existing tank farm (Arnett 1994). It is reasonable to expect that, following the tank failure accident, drinking water at the ICPP would be closely monitored, and that appropriate treatment or substitution of the drinking water supply would occur. Exposure for INEL personnel through ingestion of contaminated drinking water is therefore neglected.

- A member of the public is assumed to reside at Atomic City on the southern boundary of the INEL, approximately 18 kilometers (11 miles) from the ICPP. This person and this person's land are directly downgradient of the ICPP when the peak concentrations in the groundwater plume arrive. This individual obtains 100 percent of the drinking water from a well at this location. For the purposes of this analysis, no interdiction or evacuation take place for the maximally exposed individual.

Radiological exposures to the radioactive constituents present in this analysis were calculated by Arnett (1994). Infiltration to the aquifer would occur over approximately 200 years, and the concentration of the limiting radionuclide, strontium-90, would reach a peak concentration of 2 picocuries per liter 300 years after tank rupture. The current drinking water standard for strontium-90 is 8 picocuries per liter.

4.1.2.4.4 Preventive and Mitigative Measures—The following preventive or mitigative measures apply to the high-level waste storage tanks at ICPP:

- Stainless-steel construction

- Secondary concrete confinement vault

- Instrumentation:

  - Tank liquid level

  - Vault sump liquid level

Table 4.1.3-1. Beyond design basis accident for high-level waste.

Accident[a]	Annual frequency	Dose to MEI[b] (rem)	Risk of fatal cancer per year		
			MEI	Population, 50% meteorology	Population, 95% meteorology
Aircraft crash into calcine bin set	2×10^{-7}	1.1	1.1×10^{-10}	1.9×10^{-7} (9.5×10^{-1})	1.0×10^{-6} (5.0)

a. Fatal cancer risk = dose × accident frequency × (5.0×10^{-4} fatal cancers per rem). Number in parentheses indicates total number of fatal cancers in the population if the accident occurs.
b. MEI - maximally exposed individual.

Standard postulated accidents identified in the literature to be considered for radioactive solid waste facilities include confinement failure, fires and explosions, severe natural phenomena (such as earthquakes) and offsite manmade hazards (such as aircraft crashes) (Brynda et al. 1986). The two scenarios selected for the CSSFs (the aircraft crash in this section and the earthquake in Section 4.1.2.3) incorporate conservative aspects of all these accidents.

To immobilize high-level radioactivity, the liquid wastes are calcined to granular solids and fines at the ICPP. These granular solids are also highly radioactive, but they are chemically and physically more stable than the radioactive liquids. While nuclear fuels will no longer be reprocessed at the ICPP [except under Alternative D analyzed in the EIS (DOE 1995)], a large quantity of liquid waste from prior operations is stored in underground storage tanks and will be calcined. The calcined solids are pneumatically transferred from the calcining facility to nearby CSSFs, which consist of stainless-steel storage bins enclosed in seven separate reinforced-concrete vaults. The CSSFs provide interim storage (up to several hundred years) for the calcined solids.

Each vault, a watertight structure built on lava bedrock, contains from three to seven stainless-steel storage bins bolted to the vault floor. The excavation of the vaults has been backfilled to prevent water penetration of the below-ground portions. The bins and the vaults provide the primary and secondary containment, respectively, for the radioactive calcined solids. Figure 4.1.3.1-1 shows the general design of the Fifth CSSF, which has the highest inventory of radioactive nuclides among the seven CSSFs.

This aircraft crash scenario includes the following major events:

- A large-aircraft impact penetrates the CSSF vault and causes the failure of two internal bins containing calcined solids.

- A fraction of the contents from both bins is released into the vault.

- The free-fall of calcine solids from near the top of the bins results in a fraction of the spilled solids becoming airborne in the vault.

- A fire involving aviation fuel occurs in the vault and causes all airborne calcined solids to escape to the environment.

A CSSF vault could be penetrated if directly impacted by a large, commercial jet transport traveling at a speed approaching 325 kilometers per hour (200 miles per hour) (Chung 1992b). Analyses of jet aircraft crashes at specific facilities have resulted in predicted frequencies on the order of 2×10^{-7} per facility year (Lee et al. 1994, NRC 1975).

4.1.3.1.2 Development of Radioactive Source Term—The following assumptions, (a) through (f), apply to the above scenario and to the development of the resulting source term:

(a) After breaching the concrete vault, the aircraft impact has sufficient force to cause failure of two stainless-steel bins in the Fifth CSSF. The radionuclide inventory (curies) in the contents of all seven bins (1.46×10^6 kilograms) is the material at risk. The damage ratio is two-sevenths. The radionuclide content of calcine in the ICPP CSSF is shown in Table 4.1.3.1-1.

Table 4.1.3.1-1. Representative radionuclide concentrations in Idaho Chemical Processing Plant calcine[a]

Nuclide	Concentration (Ci/kg)	Nuclide	Concentration (Ci/kg)
Strontium-90	2.7	Europium-154	2.2E-02
Yttrium-90	2.7	Neptunium-237	6.8E-06
Ruthenium-106	6.5E-04	Plutonium-238	1.6E-02
Cesium-134	2.2E-02	Plutonium-239	3.4E-04
Cesium-137	2.9	Plutonium-240	2.1E-04
Barium-137m	2.8	Plutonium-241	4.7E-02
Cerium-144	0.0	Plutonium-242	2.8E-07
Praseodymium-144	0.0	Americium-241	1.5E-04
Promethium-147	4.6E-01	Americium-243	7.3E-07

a. Source: Berreth (1988).

Table 4.1.3.1-2. Total release fractions for aircraft crash at Calcined Solids Storage Facility.

Material	Damage ratio	Airborne release fraction	Respirable fraction	Leak path factor	Total
Solid calcine	2/7	6.2E-03[a]	5.0E-02[b]	1.0[c]	9.0E-05[d]

a. Based on 14 percent spill (Chung 1992b), 40 percent fines (Berreth 1988), and 11 percent of fines becoming airborne (Ballinger 1988).

b. Source: Berreth (1988).

c. A leak path factor of 1.0 is assigned for a major failure of confinement barriers.

d. Damage ratio × airborne release fraction × respirable fraction × leak path factor.

Table 4.1.3.1-3. Radiological source term for aircraft crash at Calcined Solids Storage Facility.

	Curies			Curies	
Nuclide	Total	Respirable	Nuclide	Total	Respirable
Strontium-90	6.84E+03	3.5E+02	Neptunium-237	1.74E-02	8.7E-04
Yttrium-90	6.84E+03	3.5E+02	Plutonium-238	4.17E-01	2.1E+00
Ruthenium-106	1.68E+00	8.4E-02	Plutonium-239	8.71E-01	4.4E-02
Cesium-134	5.72E+01	2.9E+00	Plutonium-240	5.41E-01	2.7E-02
Cesium-137	7.46E+03	3.7E+02	Plutonium-241	1.24E+02	6.2E+00
Barium-137m	6.88E+03	3.6E+02	Plutonium-242	7.46E-04	3.7E-05
Promethium-147	1.18E+03	5.9E+01	Americium-241	3.86E-01	1.9E-02
Europium-154	5.72E+01	2.9E+00	Americium-243	1.87E-03	9.3E-05

For this scenario, the RSAC-5 program was used to determine the dose from external and internal pathways at three receptor locations. The three receptor locations are (1) a location within the INEL controlled access zones (inside the ICPP facility area), (2) the nearest public access at U.S. 20/26, and (3) the nearest site boundary generally south of ICPP. The assumptions for these receptors are the same as most other accidents and as summarized in Section 2.1.2.7. Nearest site boundary assumptions and parameters were used in RSAC-5 to calculate individual doses beyond the nearest site boundary.

Table 4.1.3.1-5. Summary of dose calculation results for aircraft crash at the ICPP Calcined Solids Storage Facility at Idaho Chemical Processing Plant.

Receptor location	Inhalation CEDE[a] (rem)	Air immersion EDE[a] (rem)	Ground surface EDE (rem)	Ingestion CEDE (rem)	Total EDE (rem)	Likelihood of fatal cancer	Total likelihood of health effects
Facility worker (100 m)	4.1E+00	6.1E−03	1.2E−03	NA	4.1E+00	1.6E−03	2.3E−03
U.S. 20/26 (5,870 m)	2.6E−01	5.0E−04	5.0E−05	NA	3.0E−01	1.5E−04	2.2E−04
Nearest site boundary (14 km)	1.3E−01	2.6E−04	1.1E−01	8.6E−01	1.1E+00	5.5E−04	8.0E−04
Doses to maximally exposed individual in nearby communities (rem)							
Atomic City (17 km)	1.2E−01	2.2E−04	9.7E−02	7.5E−01	1.0E+00	5.0E−04	7.3E−04
Howe (24 km)	9.3E−02	1.8E−04	7.7E−02	6.0E−01	8.0E−01	4.0E−04	5.8E−04
Arco (31 km)	7.9E−02	1.5E−04	6.5E−02	5.1E−01	7.0E−01	3.5E−04	5.1E−04
Mud Lake/Terreton (49 km)	6.1E−02	1.2E−04	5.0E−02	3.9E−01	5.0E−01	2.5E−04	3.7E−04
Craters of the Moon National Monument (51 km)	5.9E−02	1.1E−04	4.9E−02	3.8E−01	5.0E−01	2.5E−04	3.7E−04
Blackfoot (62 km)	5.4E−02	1.0E−04	4.4E−02	3.4E−01	4.0E−01	2.0E−04	2.9E−04
Idaho Falls (72 km)	5.0E−02	9.5E−05	4.1E−02	3.2E−01	4.0E−01	2.0E−04	2.9E−04
Rigby (83 km)	4.6E−02	8.8E−05	3.8E−02	3.0E−01	4.0E−01	2.0E−04	2.9E−04
Maximum sector population doses within 80-km (50-mile) radius (person-rem)						Number of fatal cancers	Total number of individuals experiencing health effects
Maximum sector population 24,000 95%	8.6E+03	8.2E-01	3.5E+02	1.3E+03	1.0E+04	5.0E+00	7.3E+00
Maximum sector population 9,100 50%	1.6E+03	1.5E-01	6.9E+01	2.5E+02	1.9E+03	9.5E-01	1.4E+00

a. EDE — effective dose equivalent; CEDE — committed EDE.

5. TRANSURANIC WASTE ACCIDENTS

Section 5.1 estimates the consequences for the selected transuranic waste accidents assuming Alternative A is chosen. Section 5.2 estimates accident consequences for Alternatives B, C, and D.

5.1 Alternative A (No Action)—Transuranic Waste

Accidents involving transuranic waste are presented in this chapter. First the screening results are given (Section 5.1.1), then abnormal events and design basis accidents are discussed in Section 5.1.2 and one beyond design basis accident is discussed in Section 5.1.3.

5.1.1 Screening Results for Transuranic Waste Accidents

Accidents selected for transuranic waste during the screening process are listed below in Table 5.1.1-1. Detailed screening methodology is discussed in Appendix A (Accident Screening Methodology).

Table 5.1.1-1. Screening results for transuranic waste accidents.

Category[a]	Accident	Section[b]
Abnormal events	•Upsets with localized impacts only[c]	Not applicable
Design basis accidents	•Radioactive Waste Management Complex (RWMC) Transuranic Storage Area (TSA) explosion	Tables 5.1.2-1, 5.2-1
	•RWMC TSA seismic	(d)
	•RWMC Waste Characterization Facility vent release	(d)
	•RWMC lava flow	5.1.2.1, 5.2
	•RWMC TSA fire	Tables 5.1.2-1, 5.2-1
Beyond design basis accidents	•Aircraft crash	Tables 5.1.3-1, 5.2-1
	•RWMC external fire/explosion	(d)
	•RWMC criticality	(d)

a. Abnormal events are in the frequency range of 10^{-3} per year or greater. Design basis accidents are generally in the range of from 10^{-6} to 10^{-3} per year. Beyond design basis accidents are generally in the range of from 10^{-7} to 10^{-6} per year.

b. Section in this chapter where consequence analysis or summary is located.

c. Family of incidents involving spills, drops, seal failures, etc., that could have an impact in the immediate vicinity only.

d. Detailed analysis not provided in this report. The accident and consequences were analyzed in existing INEL safety documentation.

The facility is illustrated in Figure 5.1.2.1-1. Current radioactive waste management operations considered in this accident analysis consist of the following:

- Monitoring of mixed low-level waste and transuranic waste previously disposed in the Subsurface Disposal Area
- Disposal of contact-handled low-level waste in the Subsurface Disposal Area
- Disposal of remote-handled low-level waste in soil vaults in the Subsurface Disposal Area
- Storage of contact-handled transuranic waste in soil-covered asphalt pads at grade in the Transuranic Storage Area
- Storage of remote-handled transuranic waste in underground storage vaults in the Intermediate-Level Transuranic Storage Facility (ILTSF).

Before disposal in the Subsurface Disposal Area, some low-level waste is stored above ground in its shipping containers (termed "exposed" in this section). During preparation for storage or disposal, some contact-handled transuranic waste is above ground in the inflated Air Support Weather Shield (ASWS) buildings.

Because the mission of the RWMC facility is storage and disposal of radioactive materials, it has a large inventory of material potentially at risk of release. Much of this material is buried and is covered with a minimum of 1 meter (3 feet) of soil. In the accident analysis conducted for this document, release mechanisms were evaluated for two broad categories of waste at RWMC: exposed waste and buried waste.

Waste material stored above ground in the Air Support Weather Shield buildings or waste awaiting burial in the Subsurface Disposal Area is subject to a wider range of release mechanisms than soil-covered/buried waste. Potential release mechanisms include the following:

- Natural events such as tornados, earthquakes, volcanism, and floods
- Operational accidents such as vehicle or aircraft crashes, fires, explosions, and spills.

Waste covered with soil or buried beneath the surface is inherently less likely to have large airborne releases. If not stored above ground and covered with 1 meter (3 feet) of soil, it has been disposed of in numerous pits, trenches, and vaults that are separated by distance and by soil "partitions." Only events capable of disturbing the soil surface and disrupting the waste are perceived to result in significant airborne releases. Such mechanisms include the following:

- Violent or unusual natural events such as earthquakes with ground rupture, volcanism, or severe flood with surface soil erosion

- Accidents occurring during major environmental restoration activities or retrieval conducted in the buried or covered waste.

All of the human-caused accident scenarios (aircraft impact, fire, etc.) perceived at RWMC involve only localized waste storage areas. Natural phenomena events such as earthquake, flood, tornado, or volcanic activity could encompass all material at risk at RWMC. These conclusions are supported by the summary of previously published RWMC accident scenarios in Table 5.1.2.1-1.

Of the natural phenomena credible for the facility, a volcanic event could result in the largest release of radionuclides from the RWMC. Of all direct volcanic hazards (lava flows, tephra eruptions, pyroclastic surges, gas emissions), lava flows have occurred most frequently on the Snake River Plain, and are the greatest hazard to property (Hackett et al. 1987). Consistent with previous analyses, (DOE-ID 1979, DOE 1979), the bounding single release mechanism for the RWMC is therefore considered to be a molten lava flow into the facility, resulting in burning of stored waste and heating and off-gassing from buried waste, causing release of radionuclides and toxic materials to the atmosphere.

Volcanology and volcanic hazards at the INEL are discussed in Section 4.6 of Volume 2 of the EIS (DOE 1995) and in Estes et al. (1995). Based on geological assessments, either of two volcanic zones could become active and result in lava flows at the RWMC. These zones are the Axial Volcanic Zone and the Arco Volcanic Rift Zone. The likelihood of volcanic activity somewhere along these zones is 1.2×10^{-4} per year and 6.7×10^{-5} per year, respectively. The combined likelihood of activity somewhere along one of these zones is about 2×10^{-4} per year. Given a lava flow along one of the volcanic zones, the conditional likelihood of lava intrusion at the RWMC is estimated to be no greater than 0.1. Therefore, the likelihood of lava intrusion at the RWMC is estimated to be 2×10^{-5} per year. This result is consistent with RWMC deep borehole data that illustrate historic recurrence intervals of 45,000 years or a likelihood of 2.2×10^{-5} per year.

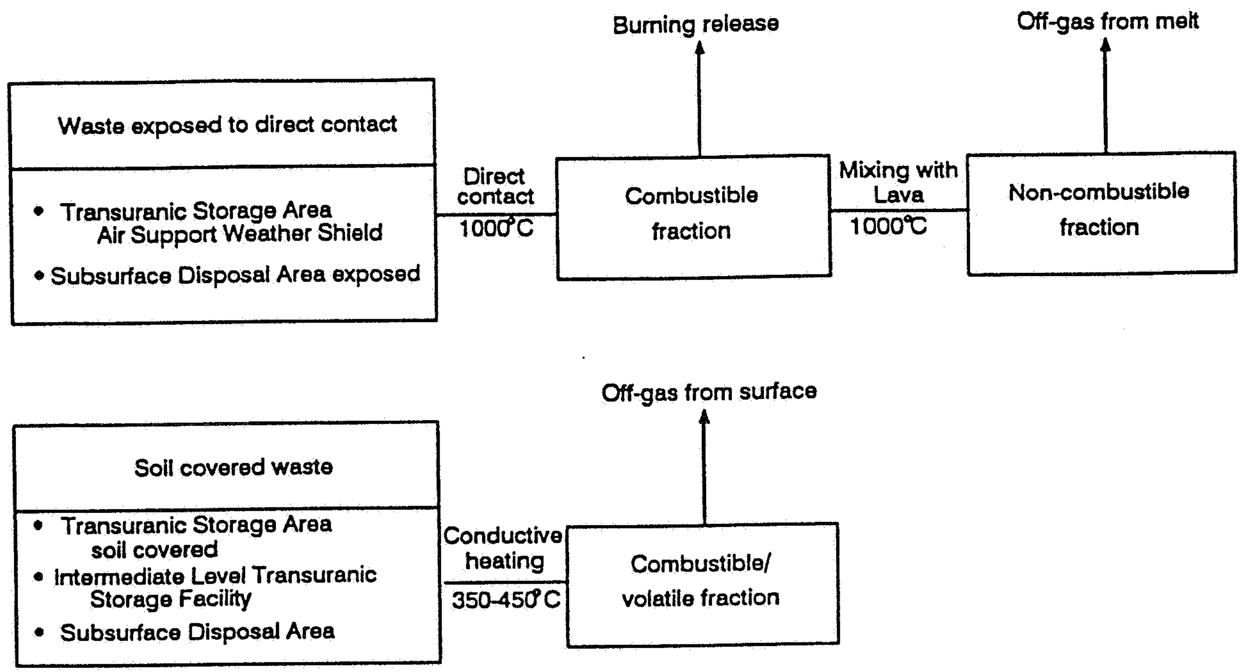

Figure 5.1.2.1-2. Postulated release scenario for lava flow at Radioactive Waste Management Complex.

The current waste volumes by accident area and vulnerable waste type are summarized in Table 5.1.2.1-2. The inventory is based on the current inventory for the Radioactive Waste Management Complex, and does not account for either resumption of shipments of transuranic wastes to INEL, or shipment of wastes offsite to the Waste Isolation Pilot Plant.

The concentrations and material at risk by isotope, and by waste location and type are summarized in Table 5.1.2.1-3.

The radionuclide concentrations were derived by correcting the concentrations in the current RWMC safety analysis report (EG&G Idaho 1986) for decay using RSAC-5. This correction is negligible for the transuranic portion of the wastes.

The various multipliers that constitute the overall release fraction and convert the material at risk to the source term are listed in Table 5.1.2.1-4. The damage ratio depends on a combination of two factors: (1) whether the waste is exposed or not, and (2) whether the waste is combustible or not. For exposed waste, the combined damage ratio is one; for transuranics at the Air Support Weather Shield buildings at the Transuranic Storage Area waste, the ratio is 0.25 combustible (EG&G Idaho 1986) and 0.75 noncombustible; for nontransuranic wastes in the Subsurface Disposal Area, these components are 0.2 combustible (EG&G Idaho 1986) and 0.8 noncombustible. For buried/soil-covered waste, the noncombustible material is unaffected so only the 0.25 damage ratio for transuranic waste and 0.2 for nontransuranic apply.

The airborne release fractions for transuranic and nontransuranic fire release, and for transuranic and nontransuranic normal off-gas releases from a molten mixture situation are as follows:

- The transuranic airborne release fraction of 5.0×10^{-4} is widely used for the burning of contaminated wastes, based on experiments performed at Pacific Northwest Laboratories (Mishima and Schwendiman 1973).
- The airborne release fraction for nontransuranic and low-level waste is assumed to be 1.0×10^{-2}. This value is consistent with generic recommendations (Elder et al. 1986) for nonvolatile solids (see Table 2.1-19).
- Transuranic and nontransuranic normal off-gas release fractions are based on engineering judgment applied to release rates measured during startup of nonradioactive in situ vitrification tests (Buelt et al. 1984).

The airborne respirable fraction is assumed to be 0.5 for a fire release of transuranic wastes based on experiments (Mishima and Schwendiman 1973). In the absence of particle size

Table 5.1.2.1-3. (continued).

Radionuclide	Concentration (Ci/m^3)	Material at risk (Ci)
Subsurface Disposal Area - Buried TRU Waste		
Nickel-59	2.11E-02	1.50E+03
Cobalt-60	2.35E-02	1.67E+03
Strontium-90	6.73E-03	4.78E+02
Yttrium-90	6.75E-03	4.79E+02
Cesium-137	1.04E-02	7.38E+02
Barium-137m	9.79E-03	6.95E+02
Uranium-238	9.58E-04	6.80E+01
Plutonium-238	6.15-03	4.37E+02
Plutonium-239	2.82E-01	2.00E+04
Plutonium-240	6.46E-02	4.59E+03
Plutonium-241	5.54E-01	3.93E+04
Americium-241	6.49E-01	4.61E+04
Subsurface Disposal Area - Exposed Low-Level Waste Awaiting Burial		
Manganese-54	2.38E-05	1.81E-02
Nickel-59	9.41E-02	7.16E+01
Cobalt-60	3.76E-01	2.86E+02
Strontium-90	7.06-02	5.37E+01
Yttrium-90	7.06-02	5.37E+01
Cesium-137	1.63E-01	1.24E+02
Barium-137m	1.54-01	1.17E+02

or other experimental data, the respirable fraction for a fire release of low-level or nontransuranic waste is assumed to be 1.0. Particle sizes observed during in situ vitrification tests have generally been less than 1 micron, so the respirable fraction from RWMC subsurface is also assumed as 1.0.

The leak path factor for this scenario is assigned a value of 1.0. The outer containment of the Air Support Weather Shield is assumed to be breached by the lava flow and the exposed waste at the Subsurface Disposal Area has no outer containment. The absence of containment allows the unmitigated release of 100 percent of the fire-driven and off-gas-generated airborne transuranic and nontransuranic airborne waste to the atmosphere. No credit is taken for filtering of off-gas through the lava as it covers exposed and buried waste containers.

The source terms for the RWMC were developed by multiplying the appropriate Transuranic Storage Area and Subsurface Disposal Area material at risk in Table 5.1.2.1-3 by the total release fractions in Table 5.1.2.1-4, and combining for combustion and off-gas mechanisms. The source terms are summarized in Table 5.1.2.1-5.

Table 5.1.2.1-5. Radiological source term for lava flow at Radioactive Waste Management Complex.

Radionuclide	Combustion source term (Ci)	Off-gas release source term (Ci)
Manganese-54	3.62E-05	6.57E-07
Nickel-59	1.43E-01	2.90E-03
Cobalt-60	5.72E-01	1.08E-02
Strontium-90	1.07E-01	2.05E-04
Yttrium-90	1.07E-01	2.05E-03
Cesium-137	2.48E-01	4.99E-01
Barium-137m	2.34E-01	4.75E-03
Uranium-233	1.32E-02	4.60E-05
Plutonium-238	6.06E-01	2.01E-03
Plutonium-239	2.60E-01	1.36E-03
Plutonium-240	6.31E-02	3.27E-04
Plutonium-241	9.13E-01	3.99E-03
Americium-241	7.88E-01	3.72E-03

Table 5.1.2.1-6. Meteorological/dispersion parameters used in dosimetry calculations for lava flow at Radioactive Waste Management Complex.

Parameter	Facility worker (100 m)	Nearest public access (US 20/26)	Nearest site boundary[a]
Receptor distance (m)	(b)	(b)	5,500
Release duration (min.)			
Combustion of exposed material	(b)	(b)	60
Off-gas generation	(b)	(b)	1,440

To convert meters to feet, multiply by 3.28.

a. Nearest site boundary values (except receptor distance) also used in calculations of doses beyond nearest site boundary.

b. No receptors postulated at onsite or nearest public access locations because scenario provides ample time to evacuate or shelter receptors at these locations.

The resulting 50-year TEDE (total effective dose equivalent) value of the nearest site boundary is 0.1 rem, as shown in Table 5.1.2.1-7. This value is almost entirely attributable to the CEDE (committed effective dose equivalent) of the inhalation pathway. This CEDE is in turn accumulated almost entirely from the initial 60-minute combustion release from exposed waste.

Combining the risk factors from Section 2.1.2.9 (Table 2.1-3) with the doses calculated for this scenario results in the calculated health effects shown in the last two columns of Table 5.1.2.1-7. Given the accident occurs, the likelihood that a member of the public would experience a health effect in the 50 years following the postulated accident ranges from about one in 20,000 at the nearest site boundary down to about one in 100,000 for a location about 80 kilometers (50 miles) from the accident. In the east-southeast compass sector (the worst case in the population of 7,900 within an 80-kilometer radius of the accident), no health effects would be expected to occur over the lifetime of the exposed population.

5.1.2.1.4 Preventive and Mitigative Measures—A lava flow event would cause the highest consequences at the RWMC facility. However, some features of the facility or actions by facility personnel may mitigate the consequences of such an event. These include the following:

- Subsurface waste is isolated from direct contact with lava.
- Some stored transuranic waste is covered with a thin soil layer.

- Existing dike work around the Subsurface Disposal Area may divert lava flow.

- Facility personnel may be able to divert a lava flow by constructing barriers or channels or using water to cool and solidify advancing flow fronts.

- With sufficient advance warning available, rapid removal of accessible transuranic waste from the RWMC would be authorized.

The RWMC is relatively remote from other INEL operating facilities; so accidents at co-located facilities are unexpected.

Emergency action mobilization would be the most important mitigative measure to reduce the consequences of such an event. Onsite evacuation and exclusion is already assumed to occur, so that no facility personnel or members of the public are exposed while within the boundaries of the INEL. However, no evacuation is assumed for an individual residing at the nearest site boundary. The only mitigative action assumed for this individual is to limit the ingestion of contaminated food items.

The primary conservatisms related to these measures in the RWMC bounding accident analysis are itemized below.

- Any efforts to divert a potential lava flow are assumed to fail. Diversion of lava flow may be successful in some instances (Hackett et al. 1987).

- It is assumed that no transuranic waste will be moved out of the path of a potential flow. Should a lava flow occur, time may be available to move some of the accessible stored waste from the RWMC.

- An individual assumed to reside at the nearest site boundary is not evacuated during the duration of the release. During the ensuing year, 10 percent of his food is still derived from vegetation and livestock raised at the nearest site boundary.

by approximately a factor of ten. The scaling factors used for transuranic waste accidents are summarized in Table 5.2-1.

Table 5.2-1. Transuranic waste accidents scaling factors—Alternatives B, C, and D.[a]

Accident	Alternative	Frequency scaling factor	Consequence scaling factor
Explosion at Radioactive Waste Management Complex (RWMC) Transuranic Storage Area (TSA)	B	1	1
	C	1	1
	D	1	1
Lava flow over RWMC	B	1	1.10 (10%)
	C	1	1
	D	1	1.20 (20%)
Fire in RWMC TSA	B	5	1
	C	10	1
	D	10	1
Aircraft crash at RWMC TSA	B	1	1
	C	1	1
	D	1	1

a. Multiply the frequency and consequence of the accident analyzed under Alternative A by the indicated scaling factor to obtain the risk of the accident under each alternative.

6. MIXED LOW-LEVEL AND LOW-LEVEL WASTE ACCIDENTS

Section 6.1 estimates the consequences for the selected mixed low-level and low-level waste accidents assuming Alternative A is chosen. Section 6.2 estimates accident consequences for Alternatives B, C, and D.

6.1 Alternative A (No Action)—Mixed and Low-Level Waste

Accidents involving mixed low-level and low-level waste are presented in this chapter. First the screening results are given (Section 6.1.1), then abnormal events and design basis accidents are discussed in Section 6.1.2 and one beyond design basis accident is discussed in Section 6.1.3.

6.1.1 Screening Results for Mixed and Low-Level Waste Accidents

Accidents selected for mixed low-level and low-level waste during the screening process are listed below in Table 6.1.1-1. Detailed screening methodology is discussed in Appendix A (Accident Screening Methodology).

Table 6.1.1-1. Screening results for mixed low-level and low-level waste accidents.

Category[a]	Low-level waste	Mixed low-level waste	Section[b]
Abnormal	•Waste Experimental Reduction Facility (WERF) Waste Storage Building (WWSB) fire	•WERF Waste Storage Building (WWSB) fire	Tables 6.1.2-1, 6.2-1
	•Radioactive Waste Management Complex (RWMC) Subsurface Disposal Area fire		Tables 6.1.2-1, 6.2-1
Design basis accidents	•RWMC Transuranic Storage Area (TSA) explosion	•RWMC TSA explosion	(c)
	•RWMC TSA seismic	•RWMC TSA seismic	(c)
	•WERF seismic	•WERF seismic	(c)
	•RWMC Waste Characterization Facility (WCF) vent release	•RWMC WCF vent release	(c)
	•WERF stack release	•WERF stack release	(c)
	•WERF fire/explosion	•WERF fire/explosion	(c)
	•RWMC lava flow	•RWMC lava flow	5.1.2.1
	•RWMC TSA fire	•RWMC TSA fire	(c)
Beyond design basis accidents	•Aircraft impact	•Aircraft impact	(c)
	•RWMC external fire/explosion	•RWMC external fire/ explosion	(c)
	•RWMC criticality	•RWMC criticality	(c)
		•WWSB major fire	Tables 6.1.3-1, 6.2-1

a. Abnormal events are in the frequency range of 10^{-3} per year or greater. Design basis accidents are generally in the range of from 10^{-6} to 10^{-3} per year. Beyond design basis accidents are generally in the range of from 10^{-7} to 10^{-6} per year.

b. Section in this document where consequence analysis or summary is located.

c. Detailed analysis not provided in this report. The accident and consequences were analyzed in existing INEL safety documentation.

Table 6.1.3-1. Beyond design basis accident for mixed and low-level waste.

Accident	Annual frequency	Dose to MEI[a] (rem)	Risk of fatal cancer per year: MEI	Risk of fatal cancer per year: Population, 50% meteorology	Risk of fatal cancer per year: Population, 95% meteorology
Beyond design basis fire at Waste Experimental Reduction Facility (WERF) Waste Storage Building	1×10^{-7}	1.4×10^{-2}	7.0×10^{-13}	Not calculated[b]	Not calculated

a. MEI - maximally exposed individual.

b. Population calculations not reported in the safety analysis documentation used for this accident.

The handling of mixed low-level and low-level waste under Alternatives B and C would approximately double over Alternative A. Under Alternative D storage and treatment of mixed low-level and low-level waste from the DOE complex would be centralized at the INEL and the inventory of mixed low-level and low-level waste at the INEL would increase by approximately ten-fold over that in Alternative A. On the basis of these factors, it was assumed that the frequency of a design basis fire at the Waste Experimental Reduction Facility (WERF) Waste Storage Building (WWSB) would increase by ten-fold over Alternative A. The frequency and consequences of a fire in the Radioactive Waste Management Complex (RWMC) Subsurface Disposal Area was assumed to increase by ten-fold over Alternative A. The scaling factors used for mixed low-level and low-level waste accidents are summarized in Table 6.2-1.

Table 6.2-1. Mixed and low-level waste accidents scaling factors - Alternatives B, C, and D[a]

Accident	Alternative	Frequency scaling factor	Consequence scaling factor
Fire in Radioactive Waste Management Complex Subsurface Disposal Area	B	1	1
	C	1	1
	D	10	10
Waste Experimental Reduction Facility (WERF) Waste Storage Building design basis fire	B	2	1
	C	2	1
	D	10	1
WERF Waste Storage Building beyond design basis fire	B	1	1
	C	1	1
	D	1	1

a. Multiply the frequency and consequence of the accident analyzed under Alternative A by the indicated scaling factor to obtain the risk of the accident under each alternative.

7. HAZARDOUS MATERIAL ACCIDENTS

Section 7.1 presents the consequences for the hazardous material accidents assuming Alternative A is chosen. Section 7.2 estimates accident consequences for Alternatives B, C, and D.

7.1 Alternative A (No Action)—Hazardous Materials

Accidents involving hazardous materials are described and analyzed in this section. First the screening results are given (Section 7.1.1), then abnormal events and design basis accidents are discussed in Section 7.1.2 and beyond design basis accidents in Section 7.1.3.

7.1.1 Screening Results for Hazardous Material Accidents

Accidents selected as bounding for hazardous materials during the screening process are listed below (Table 7.1.1-1). Detailed screening methodology is discussed in Appendix A (Accident Screening Methodology).

7.1.2 Abnormal Events and Design Basis Accidents for Hazardous Materials

Table 7.1.2-1 represents those accidents that were selected as the bounding scenarios (largest contributors to harmful effects to the public) within the ranges of incident probabilities of interest. The range of frequency of occurrence is from once every 10,000 years to once every 200,000 years. Accidents listed in Table 7.1.2-1 are presented in detail in the following sections.

7.1.2.1 Argonne National Laboratory-West: Chlorine Release and Sodium Hydroxide Release. This section describes the bounding reasonably foreseeable accident(s) involving toxic chemicals at Argonne National Laboratory-West (ANL-W).

7.1.2.1.1 Description of Accidents—The following two postulated accidents were selected as the bounding reasonably foreseeable events for Argonne National Laboratory-West:

1. Total release of chlorine gas from two cylinders in a storage cabinet outside of ANL 754

2. A sodium-water reaction involving drums of sodium stored in ANL 703, resulting in the release of sodium hydroxide.

Table 7.1.2-1. Abnormal events and design basis accidents for hazardous materials.

Accident	Annual Frequency	MEI[a] chemical concentration (percent of ERPG-3[b])
Central Facilities Area (CFA) chlorine release	1.0×10^{-4}	10
Handling accident involving existing quantities of sulfur dioxide at INEL Research Center (IRC)	1.0×10^{-4}	33
Lava flow over Radioactive Waste Management Complex (RWMC)	2.0×10^{-5}	Mercury: 30 Nitric acid: 6 Phosgene gas: 3
Argonne National Laboratory-West (ANL-W) chlorine release	1.0×10^{-5}	35
Nitric acid release at Idaho Chemical Processing Plant (ICPP)	1.0×10^{-5}	0.05
Chlorine release at ICPP	5.0×10^{-6}	7

a. MEI - maximally exposed individual at the nearest site boundary.

b. ERPG-3 - Emergency Response Planning Guide Level 3 (immediately dangerous to life and health).

ANL 703 is a low-hazard or general-use facility. The recommended Uniform Building Code (UBC) design wind speed for a low hazard facility is 70 miles per hour. A 95-mile per hour wind could demolish the structure and damage several drums. Rain or snow is assumed to accompany the high wind. The likelihood of a wind of 95 miles per hour at Argonne National Laboratory-West is 1×10^{-4} per year (Kennedy et al. 1990). It is estimated that 50 percent of the time, high winds would either be accompanied by rain or snow, or that a snow cover would exist. The estimated overall frequency of this initiating event is then 1×10^{-5} per year.

7.1.2.1.2 Development of Toxic Chemical Source Term—A small cabinet outside of the east side of ANL 754 contains two 68-kilogram (150-pound) chlorine gas cylinders. Sodium material at risk reflects the total maximum quantities expected to be in ANL 703 at any time. The materials at risk are as shown in Table 7.1.2.1-1. This table also lists the release fractions and source terms for these materials. Because the release fractions for both gases and liquids are assumed as 1.0, the source term is equal to the material at risk.

7.1.2.1.3 Exposure Calculations and Results—This section documents the computer modeling and results. The EPIcode™ (as described in Section 2.2) is used for toxic chemical impact calculations for the source terms identified in Table 7.1.2.1-1. The generic EPIcode™ input assumptions from Table 2.2-1 are used in this analysis, except as modified by the specific parameters listed in Table 7.1.2.1-2.

summarized in Table 7.1.2.1-3, and compared to respective Emergency Response Planning Guideline (ERPG) values.

Table 7.1.2.1-3. Summary of exposure calculation results for toxic chemical accidents at Argonne National Laboratory-West.[a]

Location	Chemical concentration (mg/m^3) Chlorine ERPG-3: 60 ERPG-2: 9 ERPG-1: 3
Facility worker (100 m)	8.9E+04 >ERPG-3
Nearest public access/nearest site boundary (5.24 km)	2.1E+01 >ERPG-2
Atomic City (21 km)	<ERPG-1
Mud Lake/Terreton (32 km)	<ERPG-1
Howe (35 km)	<ERPG-1
Blackfoot/Idaho Falls (50 km)	<ERPG-1
Arco (52 km)	<ERPG-1
Rigby (60 km)	<ERPG-1
Craters of the Moon (73 km)	<ERPG-1
Rexburg (74 km)	<ERPG-1

To convert meters to feet, multiply by 3.28; to convert kilometers to miles, multiply by 0.62.

a. ERPG — Emergency response planning guideline

Without seeking shelter or evacuation, a hypothetical worker fully exposed to the plume in the Argonne National Laboratory-West area at 100 meters (328 feet) would experience lethal effects from chlorine. Likewise, without evacuation, workers located at other Argonne facilities downwind could experience irreversible and life-threatening consequences from the exposures. A fully exposed member of the public located downwind at the nearest public access, at the nearest site boundary, or in nearby communities would not experience any effects. Additional information on toxic properties of chlorine is included below.

7.1.2.1.4 Preventive and Mitigative Measures—Argonne employees are trained for emergency situations and how to react to adverse conditions. Argonne also has specialized training for a group of individuals to serve as first response emergency personnel. In the event of a in the Subsurface Disposal Area, and are 100 percent soil-covered on the TSA-1 and TSA-R pads. No PCBs are, therefore, involved in the initial fire release.

As in the radiological accident scenario (Section 5.1.2.1), off-gassing would occur from soil-covered waste due to convective heating through the soil; however, the release would occur over weeks or months. For acute toxic chemical exposure, evaluations for this report were limited to the initial fire in waste exposed to direct contact with the lava.

Airborne release fractions are based on a study of potential accidents during restoration of Pit 9 at Radioactive Waste Management Complex (Reny et al. 1992). As summarized in Table 2.1-19, a release fraction of 1 percent is conservative for dispersible materials exposed to a fire (Elder et al. 1986). However, the beryllium, lead, cadmium, and lithium in the Transuranic Storage Area are not present in dispersible form, and beryllium has a very high melting point. A conservative release fraction for the beryllium exposed to the fire is 0.01 percent. Lead and cadmium, which have a much lower melting point, are assigned a release fraction of 0.1 percent. Mercury, which is volatile at standard temperature and pressure, and lithium, which is reactive, are conservatively assumed to be 100 percent released when exposed to fire. Nitric acid, also volatile near standard temperature and pressure, is assumed to be 100 percent released. It is conservatively assumed that all asbestos is present in dispersible form and is assigned a release fraction of 1 percent. Consistent with a lava flow, thermal buoyancy was assumed in the calculations. For this scenario a conservative temperature of 200°C was assumed as the release temperature of the constituents. Thermal buoyancy increased the effective release height of released materials to approximately 60 meters (200 feet).

The chlorinated hydrocarbons are nonflammable and will vaporize when exposed to heat or fire. However, when exposed to heat and fire, all halogenated compounds can be broken down to produce halogenated acids and in some cases much smaller concentrations of phosgene-type compounds. It is assumed that at least 89 percent of the chlorinated hydrocarbons will volatilize. Of the remaining 11 percent, it is conservatively assumed that 10 percent decompose to hydrochloric acid and 1 percent decompose to phosgene gas (Reny et al. 1992).

The respirable fraction is the fraction of the material at risk with particle sizes less than 10 micrometers that can be retained in the respiratory system following inhalation. The respirable fraction for hazardous materials is assumed to be 1.0.

The leak path factor accounts for the action of removal mechanisms that reduce the amount of hazardous material ultimately released to occupied spaces of a facility or to the environment. Because no credit is taken for filtering of off-gas through waste containers or Air Support Weather Shield buildings, the leak path factor for the scenario is assigned a value of 1.0. As defined in Section 2.1.2.4, the source term is the product of the material at risk, the damage ratio, the airborne release fraction, the respirable fraction, and the leak path factor. The source term for the scenario is summarized in Table 7.1.2.2-1.

7.1.2.2.3 Exposure Calculations and Results—This section documents the computer modeling and results. The EPIcode™ (Homann 1988) as described in Section 2.2 is used for toxic chemical impact calculations for the source terms identified in Table 7.1.2.2-1. The generic EPIcode™ input assumptions for Table 2.2-1 are used in this analysis, except as modified by the specific parameters listed in Table 7.1.2.2-2.

Table 7.1.2.2-2. Specific meteorological/dispersion parameters for toxic chemical release from Radioactive Waste Management Complex.

Meteorological/dispersion parameters	Facility worker (100 m)	Nearest public access (U.S. 20/26)	Nearest site boundary[a]
Receptor distance (m)	(b)	(b)	5,500
Release duration (min.)			
Combustion of exposed material	(b)	(b)	60
Release area radius (km)	0.1	0.1	0.1

To convert from meters to feet, multiply by 3.28. To convert kilometers to miles, multiply by 0.62.

a. Nearest site boundary values (except receptor distance) also used in calculations of doses beyond nearest site boundary.

b. No receptors postulated at onsite or nearest public access locations because scenario provides 36 hours to evacuate or shelter receptors at these locations.

Like the corresponding Radioactive Waste Management Complex radiological accident (5.1.2.1), normal interdiction measures make it unnecessary to model health effects for Radioactive Waste Management Complex workers or workers at co-located facilities who would be evacuated prior to any lava-generated hazardous materials release. However, as in Section 5.1.2.1, no credit is taken for evacuation of the hypothetical individual from the nearest site boundary.

The airborne concentrations, averaged over the duration of each exposure, were calculated by EPIcode™ for constituents listed in Table 7.1.2.2-1 at the following receptor locations: nearest site boundary and communities within a 80-kilometer (50-mile) radius of Radioactive Waste Management Complex. The airborne concentrations were compared to respective Emergency Response Planning Guideline (ERPG) values where available. ERPG values have not been derived for some constituents in the inventory. The effects of these constituents were assessed by comparison with other appropriate threshold values for toxic effects, including threshold limit value, time-weighted average (TLV-TWA) for ERPG-1, level of concern (LOC) for ERPG-2, and immediately dangerous to life or health (IDLH) for ERPG-3 (see Section 2.2.2 for definition of these terms).

Table 7.1.2.2-3. Summary of exposure calculation results for lava flow scenario at the Radioactive Waste Management Complex.

Location	Chemical concentration[a] (mg/m^3)					
	Mercury[b] ERPG-3: 10 ERPG-2: 1 ERPG-1: 0.01	Lithium[b] ERPG-3: 55 ERPG-2: 5.5 ERPG-1: 0.025	Hydrochloric Acid ERPG-3: 150 ERPG-2: 30 ERPG-1: 4.5	Nitric Acid[b] ERPG-3: 260 ERPG-2: 26 ERPG-1: 5.2	Asbestos[c]	Phosgene ERPG3: 4 ERPG2: 0.8 ERPG1: NA[d]
Nearest site boundary (5.5km)	3.0E+00 >ERPG-2	1.4E+00 >ERPG-1	Less than ERPG-1	1.6E+01 >ERPG-1	2.1E-02	1.3E-01
Atomic City (22 km)	2.0E+00 >ERPG-2	5.2E-01 >ERPG-1		1.1E+01 >ERPG-1	8.0E-03	9.1E-02
Arco (24 km)	1.9E+00 >ERPG-2	4.7E-01 >ERPG-1		1.0E+01 >ERPG-1	7.3E-03	8.7E-02
Howe (31 km)	1.7E+00 >ERPG-2	3.3E-01 >ERPG-1		9.0E+00 >ERPG-1	5.1E-03	7.5E-02
Craters of the Moon National Monument (39 km)	1.4E+00 >ERPG-2	1.7E-01 >ERPG-1		7.6E+00 >ERPG-1	2.7E-03	6.3E-02
Mud Lake/Terreton (62 km)	1.0E+00 =ERPG-2	4.8E-02 >ERPG-1		5.4E+00 >ERPG-1	7.5E-04	4.5E-02
Blackfoot (67 km)	9.4E-01 >ERPG-1	3.9E-02 >ERPG-1		Less than ERPG-1	6.1E-04	4.2E-02

a. When the calculated concentration is between two Emergency Response Planning Guideline (ERPG) values, the lower ERPG level is indicated; i.e., ">ERPG-2" indicates that the concentration is greater than ERPG-2, but less than ERPG-3.

b. American Industrial Hygiene Association (AIHA) ERPG values have not been established for these materials. The values reported are immediately dangerous to life or health (IDLH) for ERPG-3, 10 percent of IDLH for ERPG-2, and threshold limit value/time-weighted average (TLV/TWA) for ERPG-1.

c. Values shown are fibers per cubic centimeter, using a conversion factor of 0.03 mg/m^3 per $fiber/cm^3$. No ERPG threshold has been established for asbestos. The occupational threshold for airborne asbestos is 2 fibers per cubic centimeter (TOXnet 1993).

d. NA - not applicable. No ERPG-1 has been established for phosgene.

7.1.2.3 Central Facilities Area: Hazardous Waste Storage Facility Fire and Sewage Treatment Plant Chlorine Release. This section describes the bounding reasonably foreseeable accident(s) involving hazardous materials at the Central Facilities Area.

7.1.2.3.1 Description of Accident—The following accidents were selected as the bounding reasonably foreseeable events for the Central Facilities Area:

1. Release of chlorine gas used for treatment at the Sewage Treatment Plant (STP)

2. Sulfuric acid spilled during a fire in the Hazardous Waste Storage Facility.

The selection of these accidents is based on previous analysis (DOE 1992c; EG&G Idaho 1990b, 1992b, and 1993b) and on further screening done for this report. In this screening, the toxic/hazardous material inventories of all Central Facilities Area facilities (existing or proposed under Alternatives A or B) were compared to the 29 CFR 1910.119 threshold quantities (TQs) and the 40 CFR 355 Appendix A threshold planning quantities (TPQs). Materials not on these lists or inventories below the TQs and TPQs were dismissed from further consideration, and facilities were excluded if they did not contain materials in quantities that exceed the TQs or TPQs. The CFA-625 Engineering Research and Applications Laboratory and the Radiological and Environmental Sciences Laboratory (CFA-690 and CFA-676) were specifically examined in detail to demonstrate that their inventories do not exceed TQs/TPQs.[a] Likewise, the Central Laundry and Respirator Facility (CLRF, CFA-617) does not contain material quantities in excess of the TPQs or TQs (EG&G Idaho 1993b).

Only the Hazardous Waste Storage Facility (CFA-637) and the Sewage Treatment Plant (CFA-691) contain inventories of potentially hazardous material that exceed the TQs/TPQs (Table 7.1.2.3-1). Therefore, releases from the Sewage Treatment Plant and the Hazardous Waste Storage Facility are the only accidents evaluated further.

The Sewage Treatment Plant chlorine release could result from a number of cylinder accidents including cylinder toppling, vehicle impact, piping rupture, inadvertent valve opening, etc. Any of these events can result in the total release of chlorine. Because of the large number

a. Only one of the materials in the Engineering Research and Application Laboratory, chromium (chromic) chloride, approaches the TQs or TPQs. Its reasonably foreseeable inventory of 500 grams is about equal to the 1 pound TPQ for chromic chloride. A 5-gram release (based on a 1 percent release fraction, Class F at 0.5 meter per second) produces an airborne concentration of 0.14 milligrams per cubic meter at 100 meters (328 feet), or only 3 percent of the TWA for chromium compounds. Because of this low concentration for the only material approaching the TPQs/TQs, the Engineering Research and Application Laboratory is dismissed from further consideration.

According to a study of fire frequencies at similar facilities, the frequency of damaging fires at storage facilities is on the order of 5×10^{-3} per year (Ganti and Krasner 1984). This fire frequency estimate addresses implicitly accidents with container breaches and ignition sources. The probability of a secondary explosion resulting in a wall failure is estimated to be about 0.01. Therefore, the Central Facilities Area/Hazardous Waste Storage Facility chemical release accident is estimated to be 5×10^{-5} per year.

7.1.2.3.2 Development of Scenario and Source Term—This section describes the analysis and assumptions made to arrive at the fraction of the materials released in the accident scenarios described in the previous section.

The Sewage Treatment Plant contains one 45-kilogram (150-pound) chlorine gas cylinder for sewage treatment. The quantities of material at risk at the Hazardous Waste Storage Facility for this accident reflect the total maximum quantities expected to be in the facility at any given time. The material at risk for the Sewage Treatment Plant and the Hazardous Waste Storage Facility are as shown in Table 7.1.2.3-1 and transferred to Table 7.1.2.3-2. This table also lists the release fractions and source terms for these materials. Because the release fractions for both gases and liquids are assumed as one, the source term is equal to the material at risk except for phosgene, a product of the scenario.

Table 7.1.2.3-2. Source terms for toxic chemical accidents at Central Facilities Area.

Hazardous material	Material at risk	Release fractions[a]	Source term
	Sewage Treatment Facility		
Chlorine gas	1 150-lb cylinder	1	150 lb
	Hazardous Waste Storage Facility		
Toluene	2 55-gal drums	1	110 gal
Xylene	2 55-gal drums	1	110 gal
Sulfuric acid	12 55-gal drums (@ 94%)	1	660 gal
Nitric acid	12 55-gal drums (@ 60%)	1	660 gal
Phosgene	20 55-gal drums of chlorinated hydrocarbons	0.0119[b]	13.1 gal

To convert from gallons to liters, multiply by 3.785. To convert from pounds to kilograms, multiply by 0.454.

a. Includes leak path factor of 1.0 (i.e., no building retention, plate-out on walls, or other physical removal processes).

b. One percent of the chlorinated hydrocarbon inventory is assumed to be converted to phosgene. The phosgene molecular conversion ratio for chlorinated hydrocarbons is about 1.19 (Reny et al. 1992). Therefore the release fraction is reported as 0.0119 (0.01×1.19).

Table 7.1.2.3-3. Specific meteorological/dispersion parameters for toxic chemical releases at Central Facility Area.

Meteorological/dispersion parameters	Facility worker	Nearest public access	Nearest site boundary[a]
Receptor distance[b] (m)	100	1850	7820
Wind velocity[c] (m/s)	See Table 7.1.2.3-5 for all locations		
Release elevation (m)	EPIcode™ for all locations		
Wind stability class[c]	See Table 7.1.2.3-5 for all locations		
Release duration[d] (min.)	See Table 7.1.2.3-4 for all locations		

To convert from meter to feet, multiply by 3.28.

a. Nearest site boundary values (except receptor distance) also used for concentration calculations beyond nearest site boundary.

b. Source: Section 2.1.2.7 (Table 2.1-2).

c. For Hazardous Waste Storage Facility chemicals only; Sewage Treatment Plant chlorine releases based on generic EIS (DOE 1995) wind assumptions.

d. Instantaneous for Sewage Treatment Plant chlorine; Hazardous Waste Storage Facility chemicals calculated by EPIcode™ based on chemical's boiling point.

Table 7.1.2.3-4. Calculated release rates for toxic chemicals at Hazardous Waste Storage Facility.

Hazardous material	Boiling point (°C)	Release rate (gal/min.)	Release duration (min.)
Toluene	110.6	1.1E+01	10.2
Xylene	138	1.1E+01	10
Sulfuric acid	340	5.6E-01	198
Nitric acid	88.9	3.1E+01	21.2
Phosgene	40[a]	1.E+00	10

a. The boiling point of methylene chloride, representative of the chemical sources for phosgene.

The initial temperature of the evolved vapor plume is assumed to be the same as the temperature of the pool or the specific chemical's boiling point. Intense fires, such as the one postulated for the Hazardous Waste Storage Facility, produce high-temperature plumes that are less dense than the ambient air. The resulting buoyancy effects increase effective release heights, and may cause plumes to rise over nearby receptors during stable meteorological conditions with low wind velocities. The default assumption of stability class F with wind speeds of less than or equal to 2 meters per second may not always result in worst-case exposure estimates when the scenario indicates that elevated releases should be considered.

Table 7.1.2.3-6. Summary of exposure calculation results for toxic chemical releases at Central Facilities Area.

Location	Chemical concentration[a] (mg/m3)			
	Sewage Treatment Plant Chlorine Gas ERPG-3: 60 ERPG-2: 9 ERPG-1: 3 LC_{LO}[e]- 500	Hazardous Waste Storage Facility (HWSF) Sulfuric Acid ERPG-3: 30 ERPG-2: 10 ERPG-1: 2	HWSF Nitric Acid[b,c] ERPG-3: 260 ERPG-2: 26 ERPG-1: 5.2	HWSF Phosgene ERPG-3: 4 ERPG-2: 0.8 ERPG-1: NA[d]
Facility worker (100 m)	4.5E+04 >ERPG-3 > LC_{LO}	3.5E+00 >ERPG-1	7.6E+01 >ERPG-2	1.1E+01 >ERPG-3
Nearest public access (1.85 km)	5.8E+01 >ERPG-2	3.1E+00 >ERPG-1	3.3E+01 >ERPG-2	1.7E+00 >ERPG-2
Nearest co-located facility (4.7 km)	1.3E+01 >ERPG-2	1.4E+00 >ERPG-1	1.7E+01 >ERPG-1	6.5E-01
Nearest site boundary (7.82 km)	5.8E+00 >ERPG-1	Less than ERPG-1	1.0E+01 >ERPG-1	N/A
Atomic City (18 km)	Less than ERPG-1		Less than ERPG-1	
Howe (28 km)				
Arco (31 km)				
Craters of the Moon (49 km)				
Mud Lake/Terreton (52 km)				
Blackfoot (60 km)				
Idaho Falls (72 km)				

a. When the calculated concentration is between the two Emergency Response Planning Guideline (ERPG) values, the lower ERPG level is shown; e.g., ">ERPG-2" indicates the concentration is greater than ERPG-2, but less than ERPG-3.

b. American Industrial Hygiene Association (AIHA) ERPG values have not been established for these materials. The values reported are IDLH for ERPG-3, 10 percent of Immediately Dangerous to Life or Health (IDLH) for ERPG-2, and threshold limit value/time-weighted average (TLV/TWA) for ERPG-1.

c. These are draft ERPGs for nitric acid per Weitzman (1992).

d. NA - not appropriate. No ERPG-1 has been established for phosgene.

e. LC_{LO} (lower concentration limit for lethality) for a 5-minute exposure has been reported by National Institute of Occupational Safety and Health (NIOSH).

***Nitric acid** is toxic and highly corrosive. It can be corrosive to the skin, eyes, nose, mucous membranes, respiratory tract, or other tissue. Low concentrations are mildly irritating. Higher concentrations if inhaled can cause severe pulmonary distress, and death. Nitric acid can decompose when heated, to form highly toxic fumes of nitric oxide and hydrogen nitrate. Concentrations of 260 to 390 milligrams per cubic meter are dangerous for short exposures of 30 to 60 min. Concentrations of 520 to 1,800 milligrams per cubic meter may be fatal after even very short exposures (TOXnet 1993).*

***Phosgene**, also known as carbonyl chloride, is a highly toxic, corrosive liquid with a low boiling point. It is toxic from intakes by inhalation, ingestion and dermal absorption. Effects from exposure may include contact burns to the skin and eyes, shortness of breath, chest pain, severe pulmonary edema, and death. At low vapor concentrations, it smells like musty hay. At higher concentrations, it has a sharp and pungent odor. It is a severe irritant to the eyes and respiratory tract and can be fatal if inhaled, even for short durations and low concentrations. Exposure to 12 milligrams per cubic meter can result in immediate irritation of the respiratory tract. 80 milligrams per cubic meter may cause lung injuries within two minutes, 100 milligrams per cubic meter for as little as 30 minutes is very dangerous, and 360 milligrams per cubic meter is rapidly fatal for exposures of 30 minutes or less (TOXnet 1993).*

7.1.2.3.4 Preventive and Mitigative Measures—The Hazardous Waste Storage Facility utilizes the standard preventive and mitigative measures as required for a facility permitted under the Resource Conservation and Recovery Act (e.g., segregation of flammables, berming, emergency response training). In the event of a major release, Central Facilities Area personnel (and other INEL personnel if appropriate) may be evacuated.

7.1.2.4 Idaho Chemical Processing Plant: Chlorine Gas Release. This section describes the bounding reasonably foreseeable accident involving toxic chemicals at the Idaho Chemical Processing Plant (ICCP).

7.1.2.4.1 Description of Accident—Release of chlorine gas at the water treatment facility was determined to be the bounding reasonably foreseeable accident at the ICPP.

The selection of this accident is based on previous analysis (Rood 1991) and on further screening done for this report. Accidental releases at the ICPP of anhydrous ammonia, hydrofluoric acid, ammonium hydroxide, hexone, nitrogen, nitric acid, propane, gasoline, and chlorine were analyzed for their potential impacts at the Advanced Test Reactor 2.6 kilometers (1.6 miles) away (Rood 1991); only anhydrous ammonia and hydrofluoric acid releases caused concentrations that exceeded the Short Term Exposure Limit (TLV-STEL). Further screening done for this document showed that a bulk chlorine release would also produce concentrations exceeding several health guidelines.

The NO_X Abatement Facility (CPP 1670), which would have contained the bulk inventory of anhydrous ammonia, will not be constructed. Thus, the current chemical accident analysis has eliminated an anhydrous ammonia spill from further consideration.

2.54-centimeter (1-inch) diameter drain valve fails on both tanks simultaneously releasing the entire inventory of chlorine.

The highest chlorine concentrations at the receptor locations will result from the largest release over the shortest time period. Chlorine in a pressurized container consists of a volume of gas over a quantity of liquid, the container pressure is dependent on the vapor pressure for chlorine (5.075×10^5 Pa). The breach of the drain line will result in the nearly instantaneous release of the gaseous component followed by the release of the remainder of the contents as the liquid boils. The release duration was assumed to be approximately five minutes.

7.1.2.4.3 Exposure Calculations and Results—Since chlorine has a vapor density of 2.44 (density of air is 1), the vapor cloud will tend to hug the ground for a time when first released and may even follow terrain in directions across or against wind directions on certain boundaries. However, as the gas becomes more diluted with air, it will at some point begin to behave like a mixture with a vapor density close to that of air. Therefore, consideration of heavy gas dispersion phenomenon is more important for higher concentrations near the source. Typical Gaussian computer codes are not capable of modeling the dispersion of heavy gases accurately. However, for most of the receptor distances of interest mixing will have greatly reduced the effect of heavy gas on dispersion. As a result the EPIcode™ should provide acceptable results for this analysis.

The primary input parameters for the chlorine release analysis are provided in Table 7.1.2.4-1. The Case 2 meteorology assumptions identified in this table are used here for the following reasons:

- Higher wind velocities can result in higher concentrations at distant receptors for dense gas releases. This is not the case for neutrally buoyant releases.

- The 4.5 meter per second, Class D meteorology conditions are currently being proposed in DOE-STD-1027 (DOE 1992d) as an alternative to the unfavorable meteorological conditions historically used for radiological accident analysis.

Results for the chlorine release are summarized in Table 7.1.2.4-2 for both adverse and nominal meteorological conditions and compared to Emergency Response Planning Guideline (ERPG) values. For the 95 percent meteorology case, without seeking shelter or evacuation, workers 100 meters downwind of the release point would experience lethal effects. A hypothetical member of the public located downwind at the nearest public access could experience adverse effects. Members of the public located at the nearest site boundary, or in nearby communities would not be expected to experience adverse impacts. For nominal meteorology conditions, workers 100 meters downwind of the release could experience lethal effects if sheltering or evacuation did not occur.

Table 7.1.2.4-2. Summary of EPIcode™ calculation results for chlorine release at Idaho Chemical Processing Plant.

	Chlorine concentration[a] (mg/m^3) ERPG-3: 6.0E+01 ERPG-2: 9.0E+00 ERPG-1: 3.0E+00	
Location	Case 1[b]	Case 2[c]
Facility worker (100 m)	8.0E+04[d] >ERPG-3	1.6E+03[d] >ERPG-3
Co-located facility (Test Reactor Area at 2.6 km)	6.0E+01 >ERPG-3	5.2E+00 >ERPG-2
Nearest public access (5.26 km)	1.9E+01 >ERPG-2	1.8E+00 >ERPG-1
Nearest site boundary (14 km)	4.0E+00 >ERPG-1	4.1E-01 <ERPG-1
Atomic City (17 km)	<ERPG-1	<ERPG-1
Howe (24 km)	<ERPG-1	<ERPG-1
Arco (31 km)	<ERPG-1	<ERPG-1
Craters of the Moon National Monument (51 km)	<ERPG-1	<ERPG-1
Blackfoot (62 km)	<ERPG-1	<ERPG-1
Idaho Falls (72 km)	<ERPG-1	<ERPG-1

To convert meters to feet, multiply by 3.28; to convert kilometers to miles, multiply by 0.62.

a. When the calculated concentration is between two Emergency Response Planning Guideline (ERPG) values, the lower ERPG level is indicated; e.g., ">ERPG-2" indicates that the concentration is greater than ERPG-2, but less than ERPG-3.

b. Case 1 is based on unfavorable meteorology, Class F meteorology with 0.5 m/s wind speed for receptors within 2 km of the release and 2 m/s for receptors beyond 2 km of the release.

c. Case 2 is based on more typical meteorology, Class D meteorology with 4.5 m/s wind speed for all receptors.

d. A lethal concentration (LC_{LO}) for a five-minute exposure to chlorine is reported at 1500 mg/m^3 (NIOSH 1987). This concentration exceeds the LC_{LO}.

Hazards associated with the laboratory are typical of those in university campus laboratories. These hazards primarily involve potential exposures to toxic or highly toxic substances (e.g., acids, hydrocarbons metals, carcinogens, mutagens, gases, chemical compounds) from potential spills, fires, and explosions. Other industrial hazards such as burns or exposure to high magnetic fields also exist. Chemical hazards have been shown to dominate radiological hazards.

Previously considered operational accidents include spills of toxic chemicals outside the facility as could occur during delivery (EG&G 1992c). Concentrations for six separate chemical spills were calculated and compared to threshold limit values (TLVs) for short-term exposures. Chemicals modeled in the postulated spills were acetone, benzene, carbon tetrachloride, chloroform, hydrofluoric acid, and xylene. Of these, the release of hydrofluoric acid was the most severe and exceeded the short-term exposure TLV at the site boundary (80 meters, 260 feet) for 20 minutes.

In addition to spills, accidents involving fires and/or explosions with subsequent chemical reactions or leaks of toxic materials have been considered. Also considered previously was the crash of a small aircraft into the laboratory, which lies near the most common down-wind leg of a left-hand approach pattern for small aircraft at Fanning Field. The most vulnerable area appears to be the northeast corner of the laboratory where the delivery area, chemical storage areas, building switchgear, emergency diesel generator, and one of two main risers for the wet-pipe fire suppression system are located. In addition, a city electrical substation for the region is just beyond the northeast corner property lines.

The present analysis screens a series of separate and independent releases of individual toxic and highly toxic chemicals to determine the chemicals that, if accidentally released, would result in the highest potential consequences. The amount of material involved in each case considered what could theoretically be present in maximum quantities allowable by the building codes as if the chemical was the only one present in the laboratory. The accidents bound facility-wide events such as an aircraft crash because they involve the maximum quantity of toxic or highly toxic material allowable. In the absence of formal chemical acceptance criteria and rigorous inventory management, quantities of chemicals as large as the allowable values appear reasonable foreseeable in the National Environmental Protection Act (NEPA) sense. Chemicals selected for analysis are based on their toxicity and physical properties; specifically selected were chemicals listed in 40 CFR 355 (CFR 1993a) that are in the current INEL Research Center inventory system.

Accidents with the highest consequences from this screening process involve the release of sulfur dioxide, which could result from a handling or delivery accident. The likelihood of these accidents is composed of several elements: the likelihood of a handling accident initiating event; the likelihood of having nominal or maximum allowable quantities of the source term; and the likelihood that the incident involves the entire allowable quantity.

Table 7.1.2.6-1. Inventory quantities and input data for INEL Research Center.

Chemical	IRC chemical control limit[a] (kg)	Material at risk[b] (kg)	Current IRC inventory (kg)	Release fraction	Source term (kg)
Acrylamide	45.4	45.4	2.05	0.01	0.454
Aldrin	1.8	1.8	0.0001	0.01	0.018
Allyl alcohol	1.8	1.8	0.0854	1	1.8
Aniline	455	455	2.417	1	455
Boron trifluoride	45.4	45.4	0.1	1	45.4
Bromine	18.1	18.1	0.465	1	18.1
Cadmium oxide	1.8	1.8	1.45	0.01	0.018
Carbon disulfide	1725	250	0.0063	1	250
Chloroform	2727	295	46.62	1	295
Cyanogen bromide	1.8	1.8	0.025	0.5	0.9
Dimethyl sulfate	1.8	1.8	0.1	1	1.8
Endrin	1.8	1.8	0.001	0.01	0.018
Ethylene oxide	11.1	7.8	0.225	1	7.8
Ethylenediaminenene	2727	1200	2.247	1	1200
Hydrazine	1.8	1.8	0.5	1	1.8
Hydrogen sulfide	112	112	0.0017	1	112
Lindane	1.8	1.8	0.001	0.01	0.018
Mercury salts	1.8	1.8	3.7	0.01	0.018
Fuming nitric acid	18.1	18.1	3.935	1	18.1
Nitric oxide	1.5	1.5	0.019	1	1.5
Nitrobenzene	2727	492	1.225	1	492
Phenol	1.8	1.8	2.236	1	1.8
Cyanide salts	1.8	1.8	4.078	0.5	0.9
Sulfur dioxide	211	68	0.454	1	68
Sulfuric acid	91	91	82.8	1	91

a. Inventory is based on 1994 Draft IRC Chemical Management System.
b. Material is lesser quantity of either 1994 Draft IRC Chemical Management or maximum single cylinder quantity for pressurized gases available through commercial suppliers.

Table 7.1.2.6-2. Specific meteorological/dispersion parameters for toxic chemical release from accident at INEL Research Center.

Meteorological/dispersion parameters	Site boundary	Other receptors
Receptor distance (m)[a]	100	(b)
Release duration (min.)	60[c]	60[c]

To convert meters to feet, multiply by 3.28.

a. From northeast corner of INEL Research Center.

b. Distances to other receptors: 183 meters to U.S. 20 closest approach; 536 meters to DOE-ID Headquarters complex); 677 meters to A.H. Bush Elementary School; 920 meters to Engineering Research Office Building (EROB); 1.15 kilometers to Willow Creek Building; 20 kilometers to Rigby; 40 kilometers to Rexburg; 49 kilometers to Blackfoot; 50 kilometers to Terreton; 63 kilometers to Atomic City.

c. A 60-minute release was assumed for comparison to Emergency Response Planning Guideline (ERPG) limits. Additional calculations for bounding gaseous chemicals was done with a 5-minute release time.

A release of bounding inventory quantities of sulfur dioxide, hydrogen sulfide, boron trifluoride, carbon disulfide, or nitric oxide would produce offsite concentrations exceeding the ERPG-3 threshold concentration. Without evacuation, exposures at concentrations exceeding the ERPG-3 levels could potentially produce life threatening health effects. In the case of sulfur dioxide, these health effects could be observed as far away as approximately 700 meters (0.4 mile) from the release point under worst-case meteorological conditions. Humans exposed for one hour to levels exceeding ERPG-2 levels could potentially experience non-life-threatening but nonreversible health effects. Humans exposed for 1 hour to levels exceeding ERPG-1 levels could potentially experience non-life threatening and reversible health effects.

Concentrations for lethality for 5-minute exposures are reported (NIOSH 1987) for several of the candidate chemicals. Without evacuation or sheltering, a 5-minute release duration for the maximum inventory of sulfur dioxide would produce concentrations exceeding the reported 5-minute lethality threshold (LC_{LO-5}) of 7.8×10^3 milligrams per cubic meter to a distance of approximately 200 meters (650 feet) from the release point. Without evacuation or sheltering, a 5-minute release duration for the maximum inventory of hydrogen sulfide would produce concentrations exceeding the 5-minute threshold (LC_{LO-5}) of 8×10^2 milligrams per cubic meter to a distance of approximately 700 meters (0.4 mile) from the release point.

Table 7.1.2.6-4. Summary of calculated chemical concentrations based on current inventories at INEL Research Center.

Location	Chemical concentration[a] (mg/m3)					
	Sulfur Dioxide ERPG-3: 39 ERPG-2: 7.8 ERPG-1: 0.8	Hydrogen Sulfide ERPG-3: 140 ERPG-2: 42 ERPG-1: 0.14	Boron Trifluoride ERPG-3: 280 ERPG-2: 28 ERPG-1: 2.8	Carbon Disulfide ERPG-3:1550 ERPG-2: 155 ERPG-1:31	Ethylene Oxide ERPG-3: 1440 ERPG-2: 144 ERPG-1: 1.8	Nitric Oxide ERPG-3: 37 ERPG-2: 19 ERPG-1: 3
Site Boundary/Storage Company (100 m)	1.3E+01 >ERPG-2				5.2E+00 >ERPG-1	
U.S. 20 (180 m)	4.1E+00 >ERPG-2					
DOE Building (540 m) (about 300 workers)						
A.H. Bush Elementary (680 m) (about 700 pupils and teachers)						
Engineering Research Office Building (EROB) (920 m) (about 750 workers)				<ERPG-1		
Willow Creek Building (1.15 km) (about 1125 workers)						
Rigby (20 km) Rexburg (40 km) Blackfoot (49) Mud Lake/Terreton (50 km) Atomic City (63 km)						

To convert from meters to feet, multiply by 3.28. To convert from kilometers to miles, multiply by 0.621.

a. When the calculated concentration is between two Emergency Response Planning Guideline (ERPG) values, the lower ERPG level is shown; e.g., ">ERPG-2" indicates that the concentration is greater than ERPG-2, but less than ERPG-3.

Effects of building holdup and plating out of chemicals were not considered in this accident case. It is likely that for most release scenarios which do not involve total building collapse, that some credit could be taken for these factors. For a bounding accident case, however, it was postulated that the release occurred outside the building.

Several conservative meteorological and plume model assumptions were made for this analysis. For releases on the north side of the INEL Research Center complex, building wake effects would significantly disperse airborne chemicals transported in north or south directions. No credit was taken for wind shift during the release and the wind direction to transport in the direction of each receptor. No credit was taken for the existing wide distribution of chemicals throughout the INEL Research Center. For this scenario, it is assumed that the chemicals have been moved to one location for shipping to or from the complex.

The bounding release mechanism for the Calcined Solids Storage Facility is an aircraft crash into the facility. The major events in this accident scenario are as follows:

- A large aircraft crashes directly into the Fifth Calcined Solids Storage Facility.
- The impact has sufficient force to cause catastrophic failure of the vault and breach of two steel bins.
- The fuel in the aircraft is released to the facility and is ignited.
- The ensuing fire involves the spilled calcine, resulting in atmospheric release of toxic material.

7.1.3.1.2 Development of Scenario and Source Term—The toxic chemical source term is developed in Table 7.1.3.1-1 by multiplying the appropriate material at risk quantity by the damage ratio (14 percent as developed in Section 4.1.3.1.4), airborne release fractions, respirable fractions, and leak path factors.

Table 7.1.3.1-1. Source term for toxic chemical release from aircraft crash at Calcined Solids Storage Facility.

Constituent	Material at risk (kg)	Airborne release fraction[a]	Respirable fraction[b]	Leak path factor[c]	Sourceterm[d] (kg)
ZrO_2	8.28E+04	6.22E-03	5.0E-02	1.0	2.57E+01
CdO	4.16E+03	6.22E-03	5.0E-02	1.0	1.29E+00
Hg	4.16E+03	6.22E-03	5.0E-02	1.0	1.29E+00
Cr	4.16E+03	6.22E-03	5.0E-02	1.0	1.29E+00

To convert from kilograms to pounds, multiply by 2.2.

a. Based on 14 percent of calcine spilled to vault (Chung 1992b), 40 percent are fines (Berreth 1988), and 11 percent of fines become airborne (Ballinger 1988).

b. Source: Berreth (1988).

c. The aircraft crash scenario involves major failure of both primary and secondary confinement barriers. A leak path factor of 1.0 is assigned for a major failure of confinement barriers

d. Source term = material at risk × airborne release fraction × respirable fraction × leak path fraction.

downwind concentrations at the receptor locations, and corresponding ERPG-substitute values. Mercury was found to dominate potential health effects from this scenario.

Table 7.1.3.1-3. Summary of exposure calculation results for toxic chemical release from aircraft crash at Calcined Solids Storage Facility.

Location	Chemical concentration[a] (mg/m^3)			
	Zirconium[b] ERPG-3: 500 ERPG-2: 50 ERPG-1: 5	Cadmium[b] ERPG-3: 50 ERPG-2: 5 ERPG-1: 0.05	Mercury[b] ERPG-3: 10 ERPG-2: 1 ERPG-1: 0.01	Chromium[b] ERPG-3: 30 ERPG-2: 3 ERPG-1: 0.5
Facility worker (100 m)	4.1E-01 <ERPG-1	2.1E-02 <ERPG-1	2.1E-02 >ERPG-1	2.1E-02 <ERPG-1
Nearest public access (5.26 km)				
Nearest site boundary (14 km)				
Atomic City (17 km)				
Howe (24 km)	<ERPG-1 for all locations except for mercury for facility worker			
Arco (31 km)				
Craters of the Moon National Monument (51 km)				
Blackfoot (62 km)				
Idaho Falls (72 km)				

To convert from meters to feet, multiply by 3.28; to convert from kilometers to miles, multiply by 0.62.

a. When the calculated concentration is between two Emergency Response Planning Guideline (ERPG) values, the lower ERPG level is indicated; e.g., ">ERPG-2" indicates that the concentration is greater than ERPG-2, but less than ERPG-3.

b. American Industrial Hygiene Association ERPG values have not been established for these materials. The values reported are immediately dangerous to life or health (IDLH) for ERPG-3, 10 percent of IDLH for ERPG-2, and TLV/TWA for ERPG-1.

7.1.3.2 Argonne National Laboratory-West: Aircraft Crash into Hot Fuel Examination Facility (Toxic Chemical). This section describes the toxicological consequences of the bounding reasonably foreseeable accident at the Hot Fuel Examination Facility at Argonne National Laboratory-West (ANL-W).

7.1.3.2.1 Description of the Accident—A crash of a large commercial jet transport into the Hot Fuel Examination Facility was selected as the bounding reasonably foreseeable accident. The accident initiated by this event results in penetration of the Main Cell and in a fire in the facility involving aviation fuel. The radiological consequences of this accident are presented in Section 3.1.3.3. That section (in 3.1.3.3.1) provides the basis for accident selection, documents the facility and postulated accident scenarios considered, and discusses possible initiating events. A qualitative assessment of scenario likelihood results in an estimated likelihood of 1×10^{-7} per year.

The bounding release mechanism for Hot Fuel Examination Facility is an aircraft crash into the facility. The major events in this accident scenario are as follows:

- A large aircraft crashes directly into the Hot Fuel Examination Facility (Building 785).
- The impact has sufficient force to cause catastrophic failure of the building structure and breach of the Main Cell (Johnson 1993b).
- The fuel in the aircraft is released to the facility and is ignited.
- The ensuing fire involves the Waste Isolation Pilot Plant transuranic waste in the Decontamination Cell, High Bay, and Hot Repair Area, resulting in atmospheric release of toxic material.

7.1.3.2.2 Development of Source Term—The toxic constituents of the Waste Isolation Pilot Plant transuranic wastes include volatile organic compounds, and particulates consisting of elemental metals or nitrate salts, as shown in Table 7.1.3.2-1.

The toxic chemical source term is developed in Table 7.1.3.2-2 by multiplying the appropriate material at risk quantity by the damage ratio, airborne release fractions, respirable fractions, and leak path factors.

Table 7.1.3.2-2. Source term for toxic chemical release from aircraft crash at Hot Fuel Examination Facility.

Constituent	Material at risk[a] (23 drums) (g)	Airborne release fraction[b]	Source term (g)
	Volatile Organic Compounds		
1,1,1-trichloroethane	1.93E+04	0.89	1.72E+04
Carbon tetrachloride	2.08E+04	0.89	1.85E+04
1,1,2-trichloro-1,2,2-trifluoroethane	1.23E+04	0.89	1.09E+04
Trichloroethylene	1.30E+04	0.89	1.16E+04
Methylene chloride	1.33E+03	0.89	1.18E+03
Methyl alcohol	2.65E+01	1.0	2.65E+01
Butyl alcohol	9.96E+00	1.0	9.96E+00
Xylene	6.65E+01	1.0	6.65E+01
	Solids (Particulates)		
Cadmium	9.96E+00	1.0E-03	9.96E-03
Lead	2.74E+04	1.0E-03	2.74E+01
Mercury	1.18E+04	1.0	1.18E+04
Beryllium	3.66E+02	1.0E-04	3.66E-02
Asbestos	9.11E+03	1.0E-02	9.11E+01
Lithium	5.89E+03	1.0	5.89E+03
	Other		
Nitric acid	6.30E+03	1.0	6.30E+03
Nitrates	1.23E+03	1.0	1.23E+03
	Combustion Products		
Phosgene[c]	6.67E+03	1.0	6.67E+03
Hydrochloric Acid	6.67E+03	1.0	6.67E+03

To convert grams to pounds, multiply by 2.2×10^{-3}.

a. Data from Gratson (1990).

b. Data from Reny et al. (1992).

c. One percent of the mass of chlorinated hydrocarbons in the stored Waste Isolation Pilot Plant waste (6.67E+04 g) is converted to phosgene during combustion, and ten percent is converted to hydrochloric acid, based on Reny et al. (1992).

Table 7.1.3.2-4. Summary of exposure calculation results for toxic chemical release from aircraft crash at Hot Fuel Examination Facility.

Location	Chemical concentration[a] (mg/m^3)							
	TCE[b] ERPG-3: 2750 ERPG-2: 275 ERPG-1: 55	Carbon tetrachloride[b] ERPG-3: 1860 ERPG-2: 186 ERPG-1: 31	Mercury[b] ERPG-3: 10 ERPG-2: 1 ERPG-1: 0.01	Asbestos[c]	Lithium[b] ERPG-3: 55 ERPG-2: 5.5 ERPG-1: 0.025	Nitric Acid[b] ERPG-3: 260 ERPG-2: 26 ERPG-1: 5.2	Phosgene ERPG-3: 4 ERPG-2: 0.8 ERPG-1: NA[d]	HCl ERPG-3: 150 ERPG-2: 30 ERPG-1: 4.5
Facility worker (100 m)	Less than ERPG-1	Less than ERPG-1	<1.00E-04 <ERPG-1	<1.00E-08	Less than ERPG-1	Less than ERPG-1	Less than ERPG-2	Less than ERPG-1
Nearest public access/nearest site boundary (5.24 km)			2.2E-02 >ERPG-1	1.0E-04				
Atomic City (21 km)			Less than ERPG-1	6.5E-06				
Mud Lake/Terreton (32 km)				2.2E-06				
Howe (35 km)				1.4E-06				
Blackfoot/Idaho Falls (50 km)				2.5E-07				
Arco (52 km)				2.0E-07				
Rigby (60 km)				1.0E-07				
Rexburg (74 km)				3.7E-08				

a. When the calculated concentration is between two Emergency Response Planning Guideline (ERPG) values, the lower ERPG level is shown; e.g., ">ERPG-2" indicates that the concentration is greater than ERPG-2, but less than ERPG-3.
b. AIHA ERPG values have not been established for these materials. The values reported are IDLH for ERPG-3, 10 percent of immediately dangerous to life or health (IDLH) for ERPG-2, and threshold limit value/time-weighted average (TLV/TWA) for ERPG-1.
c. Values reported are fibers per cubic centimeter, using a conversion factor of 0.03 mg/m^3 per $fibers/cm^3$ (TOXnet 1993). No ERPG threshold has been established for asbestos. The occupational threshold for airborne asbestos is 2 fibers per cubic centimeter.
d. NA - not applicable. No ERPG-1 has been established for phosgene.
Abbreviations: TCE - 1,1,1-trichloroethane, HCl - hydrochloric acid.

***Nitric acid** is toxic and highly corrosive. It can be corrosive to the skin, eyes, nose, mucous membranes, respiratory tract, or other tissue. Low concentrations are mildly irritating. Higher concentrations if inhaled can cause severe pulmonary distress, and death. Nitric acid can decompose when heated, to form highly toxic fumes of nitric oxide and hydrogen nitrate. Concentrations of 260 to 390 milligrams per cubic meter are dangerous for short exposures of 30 to 60 minutes. Concentrations of 520 to 1,800 milligrams per cubic meter may be fatal after even very short exposures (TOXnet 1993).*

***Phosgene**, also known as carbonyl chloride, is a highly toxic, corrosive liquid with a low boiling point. It is toxic from intakes by inhalation, ingestion, and dermal absorption. Effects from exposure may include contact burns to the skin and eyes, shortness of breath, chest pain, severe pulmonary edema, and death. At low vapor concentrations, it smells like musty hay. At higher concentrations, it has a sharp and pungent odor. It is a severe irritant to the eyes and respiratory tract and can be fatal if inhaled, even for short durations and low concentrations. Exposure to 12 milligrams per cubic meter can result in immediate irritation of the respiratory tract. Eighty milligrams per cubic meter may cause lung injuries within two minutes, 100 milligrams per cubic meter for as little as 30 minutes is very dangerous, and 360 milligrams per cubic meter is rapidly fatal for exposures of 30 minutes or less (TOXnet 1993).*

***Hydrochloric acid** is a irritant to the respiratory tract, skin, eyes, and mucous membranes. More severe exposures result in pulmonary edema, and often laryngeal spasm. A concentration of 53 milligrams per cubic meter causes irritation of the throat after short exposure. Concentrations of 75 to 150 milligrams per cubic meter are tolerable for one hour; concentrations of 1,500 to 3,000 milligrams per cubic meter are dangerous, even for brief exposures (TOXnet 1993).*

7.1.3.2.4 Preventive and Mitigative Measures—A large aircraft crash is beyond the design basis of the Hot Fuel Examination Facility. However, design features or personnel actions at the Hot Fuel Examination Facility that may mitigate accident consequences include the following:

- A minimum staffing level is maintained, and routine inspections are conducted of the Hot Fuel Examination Facility cells and Waste Isolation Pilot Plant waste activity areas by safety personnel to ensure that administrative limits are maintained.

- Emergency response facilities and procedures are in place for ANL-W, INEL, and surrounding area. It is expected that release of radioactive materials associated with the aircraft crash would activate the INEL Take Cover response and would cause facility workers to seek cover.

- DOE Fire Departments at ANL-W and Central Facilities Area are available on short notice.

7.1.3.3 Argonne National Laboratory-West: Aircraft Crash into Fuel Cycle Facility (Toxic Chemical). This section describes the toxicological consequences of the bounding reasonably foreseeable accident at the Fuel Cycle Facility (FCF) at Argonne National Laboratory-West.

Table 7.1.3.3-1. Toxic chemical source term from aircraft crash at Fuel Cycle Facility.

Constituent	Material at risk[a] (g)	Airborne release fraction	Respirable fraction	Leak path factor[b]	Source term (g)
Cadmium	5.6E+05	1.0E-02[a]	1.0E+00	1.0E+00	5.6E+03

To convert grams to pounds, multiply by 454.

a. Source: Elder et al. (1986).

b. The aircraft crash scenario involves major failure of both primary and secondary confinement barriers. A leak path factor of 1.0 is assigned for a major failure of confinement barriers.

Table 7.1.3.3-2. Specific meteorological/dispersion parameters for toxic chemical release aircraft crash at Fuel Cycle Facility.

Meteorological/dispersion parameters	Facility worker	Nearest public access/ nearest site boundary[a]
Receptor distance (m)	100	5,240b
Release duration (min.)	60	60

To convert meters to feet, multiply by 328.

a. Nearest public access/nearest site boundary values (except receptor distance) also used for concentration calculations beyond nearest site boundary.

b. Source: Section 2.1.2.7 (Table 2.1-2).

The airborne concentrations, averaged over the duration of each exposure, were calculated by EPIcode™ for cadmium listed in Table 7.1.3.3-2, at the following receptor locations: facility worker in the ANL-W facility area, the nearest public access/nearest site boundary at U.S. Highway 20, and communities within a 80-kilometer (50-mile) radius of ANL-W. Emergency Response Planning Guideline (ERPG) values have not been derived for cadmium. The effects of the concentrations were assessed by comparison with other appropriate threshold values for toxic effects, including threshold limit value, time-weighted average (TLV-TWA) for ERPG-1, level of concern (LOC) for ERPG-2, and immediately dangerous to life or health (IDLH) for ERPG-3 (see Section 2.2.2 for definition of these terms).

7.1.3.3.4 Preventive and Mitigative Measures—A large aircraft crash is beyond the design basis of the Fuel Cycle Facility. However, design features or personnel actions at Fuel Cycle Facility that may mitigate accident consequences include the following:

- A minimum staffing level is maintained, and routine inspections are conducted of the Fuel Cycle Facility cells by safety personnel to ensure that administrative limits are maintained.

- Emergency response facilities and procedures are in place for ANL-W, INEL, and surrounding area.

- DOE Fire Departments at Argonne National Laboratory-West and Central Facilities Area are available on short notice.

7.1.3.4 Test Area North: Depleted Uranium Fires. This section describes the toxicological consequences of the bounding reasonably foreseeable toxic chemical accident at Test Area North (TAN).

7.1.3.4.1 Description of the Accident—A large fire at TAN-628 was selected as the bounding reasonably foreseeable accident. The accident, initiated by an unspecified event, results in structural failure of the building and involves the entire inventory of depleted uranium stored there.

Both in-process and waste inventories of hazardous materials exist at Test Area North as a result of past and current research projects. Ongoing activities at Test Area North include research, development, and production of armor for the M1A1 battle tank in the Specific Manufacturing Capability (SMC) program and the remote assembly, disassembly, analysis, and storage of radioactive materials and nuclear fuels at the Test Area North Hot Shop and Fuel Storage Canal. A number of smaller projects range from research into explosive detection systems (Federal Aviation Administration (FAA) Explosives Detection System)) to experimental determinations of the behavior of water in nonnuclear thermal-hydraulic systems (Water Reactor Research Test Facility).

A preliminary accident screening process identified two toxic materials at Test Area North for consideration. Significant quantities of concentrated nitric acid and depleted uranium are stored at Test Area North facilities. Nitric acid is stored in the Process Reclamation Facility, TAN-681, and depleted uranium is stored in several locations, including TAN-628 and TAN-688. In the safety analysis reports (BWI-91-003, BWI-91-008), each of these buildings has been classified as low hazard under DOE Order 5481.1B (DOE 1987a) and as category 3 facilities under DOE Order 5480.23 (DOE 1992a, B&W 1993).

Table 7.1.3.4-1. Inventories of depleted uranium in Specific Manufacturing Capability facilities.[a]

Building	Description	Quantity of depleted uranium (kg)
TAN-628	Warehouse	1.33E+06
TAN-629	Fabrication and assembly	1.82E+04
TAN-677	Warehouse	8.52E+04
TAN-679	Rolling operations	5.00E+04
TAN-681	Waste operations	4.73E+03
TAN-682	Warehouse	2.71E+05
TAN-688	Warehouse	1.36E+06
	Cargo containers	6.26E+05
	Waste operations storage	3.36E+03

To convert from kilograms to pounds, multiply by 2.2.

a. Data from B&W (1993).

Table 7.1.3.4-2. Source term for depleted uranium fire at Specific Manufacturing Capability facility.

Material	Material at risk (kg)	Airborne release fraction[a]	Respirable fraction	Leak path factor	Source term (kg)
Depleted uranium	1.33E+06	0.001	1.0	1.0	1.33E+03

To convert from kilograms to pounds, multiply by 2.2.

a. Data from Mishima et al. (1985).

7.1.3.4.3 Exposure Calculations and Results—This section documents the computer modeling and results. The EPIcode™ (Homann 1988) (as described in Section 2.2.2.1) is used for toxic chemical impact calculations for the source terms. The generic EPIcode™ input assumptions from Table 2.2-1 are used in the analysis, except as modified by the specific parameters listed in Table 7.1.3.4-3.

Table 7.1.3.4-4. Summary of exposure calculation results for depleted uranium fire at Specific Manufacturing Capability facilities.

Location	Chemical concentration[a] (mg/m3) Uranium[b] ERPG-3: 20.0 ERPG-2: 2.0 ERPG-1: 0.2
Facility worker (100 m)	< ERPG-1
Nearest public access (593 m)	< ERPG-1
Point of maximum concentration (1.0 km)	5.7 > ERPG-2
Nearest site boundary (11.2 km)	< ERPG-1 for remaining location
Mud Lake/Terreton (14 km)	
Howe (30 km)	
Atomic City (58 km)	
Arco (58 km)	
Idaho Falls (60 km)	
Rigby (62 km)	
Rexburg (74 km)	
Blackfoot (77 km)	
Craters of the Moon (83 km)	

To convert meters to feet, multiply by 3.28; to convert kilometers to miles, multiply by 0.62.

a. When the calculated concentration is between two Emergency Response Planning Guideline (ERPG) values, the lower ERPG level is shown; e.g., ">ERPG-2" indicates that the concentration is greater than ERPG-2, but less than ERPG-3.

b. American Industrial Hygiene Association ERPG values have not been established for uranium. The values reported are immediately dangerous to life or health (IDLH) for ERPG-3, 10 percent of IDLH for ERPG-2, and threshold limit value/time-weighted average (TLV/TWA) for ERPG-1.

Table 7.2-1. Hazardous material accidents scaling factors.

Accident	Alternative	Frequency scaling factor	Consequence scaling factor
Lava flow over RWMC	B	1	1
	C	1	1
	D	1	1.2 (20%)
All others	B	1	1
	C	1	1
	D	1	1

Table 7.2-2. Hazardous material accidents under Alternative D that differ from those under Alternatives A, B, and C.

Accident	Frequency	MEI chemical concentration (mg/m^3)	MEI chemical concentration (percent of ERPG-3[a])
Hydrofluoric acid spill at ICPP	1×10^{-5}	0.078	0.2
Anhydrous ammonia release at ICPP	1×10^{-6}	82	12

a. ERPG-3 - Emergency Response Planning Guide Level 3 (immediately dangerous to life and health). MEI - maximally exposed individual at the nearest site boundary.

8. ENVIRONMENTAL REMEDIATION AND DECONTAMINATION AND DECOMMISSIONING ACCIDENTS

Section 8.1 estimates the consequences for the selected environmental remediation and decontamination and decommissioning accidents assuming Alternative A is chosen. Section 8.2 estimates accident consequences for Alternatives B, C, and D.

8.1 Alternative A (No Action)—Environmental Remediation and Decontamination and Decommissioning

Accidents involving environmental remediation and decontamination and decommissioning are described and analyzed in this section. First the screening results are given (Section 8.1.1), then abnormal events and design basis accidents are discussed in Section 8.1.2 and beyond design basis accidents in Section 8.1.3.

8.1.1 Screening Results for Environmental Remediation and Decontamination and Decommissioning Waste Accidents

Accidents selected for environmental remediation and decontamination and decommissioning (D&D) activities during the screening process are listed below in Table 8.1.1-1. Detailed screening methodology is discussed in Appendix A (Accident Screening Methodology).

Table 8.1.1-1. Environmental remediation and decontamination and decommissioning accidents.

Category[a]	Decontamination and decommissioning	Remediation	Section[c]
Abnormal events	•Upsets with localized impact only[b]	•Pit 9 stack/vent release	Not applicable
Design basis accidents	•Stack vent release		(d)
	•Movement/ handling accident		(d)
			(d)
	•Fire/explosion	•Pit 9 fire/ explosion	Table 8.1.2-1
		•Pit 9 container handling outside	(d)
		•Pit 9 major fire	Table 8.1.2-1
		•Pit 9 high winds	(d)
		•Pit 9 seismic	Table 8.1.2-1
Beyond design basis accidents	•Aircraft impact	•Pit 9 aircraft impact	(d)

a. Abnormal events are in the frequency range of 10^{-3} per year or greater. Design basis accidents are generally in the range of from 10^{-6} to 10^{-3} per year. Beyond design basis accidents are generally in the range of from 10^{-7} to 10^{-6} per year.
b. Family of incidents involving spills, drops, seal failures, etc., that could have an impact in the immediate vicinity only.
c. Section in this chapter where consequence analysis or summary is located.
d. Detailed analysis not provided in this report. The accident and consequences were analyzed in existing INEL safety documentation.

9. REFERENCES

Abbott, M. L., 1990, *Hydrofluoric Acid and Anhydrous Ammonia Spills Evaporation/Air Dispersion Modeling and Exposure Assessment*, EGG-EST-8946, EG&G Idaho, Inc., Idaho Falls, Idaho, April.

Abbott, M. L., 1992, *Toxic Vapor Cloud Impacts from Accidental Releases of Anhydrous Ammonia and Nitrogen Dioxide at the ICPP NOx Abatement Facility*, EGG-GEO-10159, EG&G Idaho, Inc., Idaho Falls, Idaho, April.

Abrahamson, S., M. Bender, S. Book, C. Buncher, C. Denniston, E. Gilbert, F. Hahn, V. Hertzbert, H. Maxon, B. Scott, W. Schull, S. Thomas, 1989, "Scientific Bases for Health Effects Models, " Part II of *Health Effects Models for Nuclear Power Plant Accident Consequence Analysis*, NUREG/CR-4214, Rev. 1, SAND85-7185, Sandia National Laboratories, Albuquerque, New Mexico, May.

ACGIH (American Conference of Governmental Industrial Hygienists), 1987, *Threshold Limit Values and Biological Exposure Indices for 1987-1988*, Cincinnati, Ohio.

ACI (American Concrete Institute), 1977, "Building Code Requirements for Reinforced Concrete," Standard 318-77.

Adams, R. M., D. C. Hampton, K. R. Ferguson, E. Hylsky (coordinators), 1975, *Hot Fuel Examination Facility North Facility Safety Report, Argonne-West, Idaho National Engineering Laboratory*, ANL-7959, Argonne National Laboratory, Argonne, Illinois.

Adams, J. P. and M. L. Carboneau, 1992,, *Fission Product Behavior During ATR Transient CAN2/EQC4*, PG-T-127, EG&G Idaho, Inc., Idaho Falls, Idaho, August.

Advanced Engineering Consultants, Inc., 1991, *Seismic Analysis and Evaluation of Octagonal Waste Tank Vaults WM-180 & WM-181 at the Idaho Chemical Processing Plant*, Idaho Falls, Idaho, February.

Ammerich, M., A. Briand, J. C. Laborde, Ph. Mulcey, J. Savornin, 1988, "Contamination Releases from HEPA Filters Under High Temperature Operating Conditions," presented at 20th DOE/NRC Nuclear Air Cleaning Conference.

ANL-W (Argonne National Laboratory-West), 1988, *Preliminary Safety Analysis Report, HFEF/S Modifications Project*, Idaho Falls, Idaho, January.

ANL-W (Argonne National Laboratory-West), 1991, *Hazardous Materials Inventory Report, SARA III*, Idaho Falls, Idaho, March 7.

ANL-W (Argonne National Laboratory-West), 1993, *Final Safety Analysis Report, Fuel Cycle Facility, Argonne National Laboratory-Idaho Site, draft*, F0000-0018-AK, Rev. 00, Idaho Falls, Idaho, March.

CFR (Code of Federal Regulations), 1992c, 40 CFR 372, Ch. 1, "Toxic Chemical Release Reporting: Community Right-to-Know," Office of the Federal Register, July 1.

CFR (Code of Federal Regulations), 1992d, 40 CFR 261.33, Ch. 1, "Identification and Listing of Hazardous Waste," Office of the Federal Register, July 1.

CFR (Code of Federal Regulations), 1993a: 40 CFR 355 Appendices A and B, Lists of Extremely Hazardous Substances and Their Threshold Planning Quantities), Office of the Federal Register.

Chung, T., 1992a, *Structural Assessment Report, Liquid Waste Storage Tanks, WM-180 through WM-190, WM-100, -101, -102, and WL-101*, Science Applications International Corporation, Idaho Falls, Idaho, December 17.

Chung, T., 1992b. *Evaluation of Airplane Crashing into 5th CSSF Vault*, Science Applications International Corporation, Idaho Falls, Idaho, November 6.

Chung, T., 1993, Memorandum to B. House, regarding "Potential structural damage to other buildings caused by the collapse and impact of the ICPP Main Stack (CPP-708) during the maximum credible seismic event," Science Applications International Corporation, Idaho Falls, Idaho, February 17.

Clawson, K. L., G. E. Start, N. R. Ricks, 1989, *Climatography of the Idaho National Engineering Laboratory, 2nd Ed.*, DOE/ID-12118, U.S. Department of Energy, Idaho Operations Office, Idaho Falls, Idaho.

Croff, A. G., 1983, "ORIGEN2: A Versatile Computer Code for Calculating the Nuclide Compositions and Characteristics of Nuclear Materials," *Nuclear Technology*, 62, p. 335, September.

Dearien, J. A., 1971, *CPP Seismic Analysis Report*, DEA-15-71, EG&G Idaho, Inc., August 17.

DOE (U.S. Department of Energy), 1979, *Environmental and Other Evaluations of Alternatives for Long-Term Management of Stored INEL Transuranic Waste*, DOE/ET-0081 (Revised), Washington, D.C., February.

DOE (U.S. Department of Energy), 1986, Order 5480.1B, "Environmental Safety and Health Program for DOE Operations," Washington, D.C., September 26.

DOE (U.S. Department of Energy), 1987a, Order 5481.1B, "Safety Analysis and Review System," Washington, D.C., May 19, 1987.

DOE (U.S. Department of Energy), 1988a, *Internal Dose-Rate Conversion Factors for Calculation of Dose to the Public*, DOE/EH-0071, Washington, D.C.

DOE (U.S. Department of Energy), 1988b, *External Dose-Rate Conversion Factors for Calculation of Dose to the Public*, DOE/EH-0070, Washington, D.C.

DOE-ID (U.S. Department of Energy-Idaho Operations Office), 1979, *Environmental and Other Evaluations of Alternatives for Long-Term Management of Buried INEL Transuranic Waste*, IDO-10085, Idaho Falls, Idaho, December.

DOE-ID (U.S. Department of Energy-Idaho Operations Office), 1987, *Environmental Assessment: Fuel Processing Restoration at the Idaho National Engineering Laboratory*, DOE/EA-0306, Idaho Falls, Idaho, August.

DOE-ID (U.S. Department of Energy, Idaho Operations Office,), 1991a. *Air Permitting Handbook*, Appendix E: Ambient Air Receptors for Air Modeling at the INEL, Idaho Falls, Idaho, February.

DOE-ID (U.S. Department of Energy, Idaho Operations Office), 1991b, *Idaho National Engineering Laboratory Historical Dose Evaluation*, 2 volumes, DOE/ID-12119, Idaho Falls, Idaho, August.

DOE-ID (U.S. Department of Energy-Idaho Operations Office), 1991c, *Radioactive Waste Management Information for 1990 and Record-to-Date*, DOE/ID-10054(90), Idaho Falls, Idaho, July.

DOE-ID (U.S. Department of Energy-Idaho Operations Office), 1992, *DOE-ID Architectural Engineering Standards*, "Seismic Design," Section 0112-6, Rev. 13, Idaho Falls, Idaho, September 21.

EG&G Idaho (EG&G Idaho, Inc.), 1986, *Safety Analysis for the Radioactive Waste Management Complex at the Idaho National Engineering Laboratory*, WM-PD-86-011, Rev. 2, Idaho Falls, Idaho, July.

EG&G Idaho (EG&G Idaho, Inc.), 1990a, *Safety Analysis Report for Mixed Waste Storage Facility (MWSF)*, Idaho Falls, Idaho, August.

EG&G Idaho (EG&G Idaho, Inc.), 1990b, *Safety Analysis Report for the Hazardous Waste Storage Facility*, EGG-WM-PD-88-014, Rev. 1, Idaho Falls, Idaho, November.

EG&G Idaho (EG&G Idaho, Inc.), 1991, *TAN Hot Cell Complex Final Safety Analysis Report and Operations Safety Requirements Documents for the Hot Cells North Facilities*, Idaho Falls, Idaho, July 3.

EG&G Idaho (EG&G Idaho, Inc.), 1992a, *Preliminary Hazards Assessment of INEL Waste Management Facilities Operated by EG&G Idaho*, EGG-WMO-10459 Draft, Idaho Falls, Idaho, September.

EG&G Idaho (EG&G Idaho, Inc.), 1992b, *Safety Analysis Report for the Engineering Research and Applications Department Central Facility Area Building 625*, ERASAR-92-01-CFA, Idaho Falls, Idaho, September.

EG&G Idaho (EG&G Idaho, Inc.), 1992c, *Draft Environmental Assessment for the Expansion of the Idaho National Engineering Laboratory Research Center*, Idaho Falls, Idaho, September.

EPA/FEMA/DOT (U.S. Environmental Protection Agency, Federal Emergency Management Agency, U.S. Department of Transportation), 1987, *Technical Guidance for Hazards Analysis*, Washington, D.C., December.

EQE (EQE Engineering), 1988, *Seismic Evaluation of Waste Tank Vaults at the Idaho Chemical Processing Plant*, prepared for Westinghouse Idaho Nuclear Corporation, Idaho Falls, Idaho, November.

Estes, M., A. L. Lundahl, S. Williams, K. Fischer, 1995, *Water Resources Supporting Document for the INEL ER&WM EIS*, Engineering Design File EIS-WATR-0101, Science Applications International Corporation, Idaho Falls, Idaho, March.

Fast, E., Westinghouse Idaho Nuclear Company (WINCO), Idaho Falls, Idaho, 1989, Memorandum to G. T. Paulsen (WINCO), regarding, "CPP-603 North and Middle Basin Scrambled Fuel Studies," Fast-01-89, January 13.

FDA (U.S. Food and Drug Administration), 1982, *Accidental Radioactive Contamination of Human Food and Animal Feeds; Recommendations for State and Local Agencies*, October 22.

FEMA/EPA/DOT (Federal Emergency Management Agency)/U.S. Environmental Protection Agency/U.S. Department of Transportation), ca. 1989, *Handbook Of Chemical Hazard Analysis Procedures*, Washington, D.C.

Finicle, D. P., 1977, *Scoping Seismic Analysis of the TAN 607 Hot Shop*, RE-A-77-147, LTR 10-37, EG&G Idaho, Inc., Idaho Falls, Idaho, December 13.

FR (Federal Register), 1993, 58 FR 11, "Table 1 to §68.130", January 19.

Funk, C. W. and L. D. Jacobson, 1979, *Inventory And Characterization Of Spent LWR Fuel*, HEDL-TME 77-82, EG&G Idaho, Inc., Idaho Falls, Idaho, April.

Ganti, C. S. and L. M. Krasner, 1984, *Navy Fire Risk Management: A Methodology for Prioritizing Fire Protection Recommendations*, Factory Mutual Research, Norwood, Massachusetts, September.

GNS (General Nuclear Systems, Inc.), 1985, *Topical Safety Analysis Report For The Castor V/21 Cask Independent Spent Fuel Storage Installation (Dry Storage),*" V-21 SAR, Rev. 1, January.

Gratson, M. W., 1990, *Hazardous Stored TRU Waste Source Term for the RWMC's TSA*, Engineering Design File ENV-003, EG&G Idaho, Inc., Idaho Falls, Idaho, February 7.

Grove (Grove Engineering, Inc.), 1988, *Microshield Version 3 Manual*, Rockville, Maryland, April 12.

Hackett, W. R., R. C. Bartholomay, D. G. Disney, C. F. Hersley, L. G. Snider, N. C. Zentner, 1987, *Volcanic Hazards Assessment for the Proposed Super-Conducting Supercollider, Idaho National Engineering Laboratory, Eastern Snake River Plain, Idaho*, prepared for the Idaho Department of Commerce, U.S. Geological Society, December 14.

Markee, E. H. Jr., 1967, "A Parametric Study of Gaseous Plume Depletion by Ground Surface Adsorption," in *Proceedings of USAEC Meteorological Information Meeting,* C. A. Mawson, ed., AECL-2787, pp. 602-613.

Mishima, J. and L. C. Schwendiman, 1973, *Some Experimental Measurements in Fractional Airborne Release of Uranium (Representing Plutonium) During Burning of Contaminated Wastes,* BNWL-1730, Pacific Northwest Laboratory, Richland, Washington, April.

Mishima, J., M. A. Parkhurst, R. I. Scherpelz, D. E. Hadlock, 1985, *Potential Behavior of Depleted Uranium Penetrators under Shipping and Bulk Storage Accident Conditions*, PNL-5415, Pacific Northwest Laboratory, Richland, Washington, February.

Moffitt, W. C., Westinghouse Idaho Nuclear Company, Idaho Falls, Idaho, 1992, Letter to A. A. Pitrolo, U.S. Department of Energy-Idaho Operations Office, regarding "CPP-603 Fuel Transfer Plan," WCM-255-92, July 31.

Moore, R. E., C. F. Baes III, L. M. McDowell-Boyer, A. P. Watson, F. O. Hoffman, J. C. Pleasant, and C. W. Miller, 1979, *AIRDOS-EPA: A Computerized Methodology for Estimating Environmental Concentrations and Dose to Man from Airborne Releases of Radionuclides*, ORNL-5532, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Napier, B. A., R. A. Peloquin, D. L. Strenge, J. V. Ramsdell, 1988, *GENII - The Hanford Radiation Dosimetry Software System*, PNL-6584, Pacific Northwest Laboratory, Richland, Washington, November.

NCRP (National Council on Radiation Protection and Measurements), 1985, *Induction of Thyroid Cancer by Ionizing Radiation*, Regulatory Guide 80, Washington, D.C.

NEPA (National Environmental Policy Act), 1969, Public Law 89-665, 16 U.S.C. 4321-4361, 1976; 40 CFR 1500-1508, 10 CFR 1021; EO 11514, 11991.

NIOSH (National Institute for Occupational Safety and Health), 1987, *Registry of Toxic Effects of Chemical Substances, 1985-86 Edition,* Cincinnati, Ohio, April.

NIOSH (National Institute for Occupational Safety and Health), 1990, *Pocket Guide to Chemical Hazards,* Washington, D.C., June.

Noorani, Y., Advanced Sciences Inc., 1993, Unpublished work, database printout of January 6.

NRC (U.S. Nuclear Regulatory Commission), 1972, *Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling Water and Pressurized Water Reactors,* Regulatory Guide 1.25, Washington, D.C., March 23.

NRC (U.S. Nuclear Regulatory Commission), 1975, *Reactor Safety Study,* WASH-1400, Washington, D.C., October.

RSIC (Radiation Shielding Information Center), 1991, "ORIGEN2.1, Isotope Generation and Depletion Code Matrix Exponential Method," CCC-371, RSIC Computer Code Collection, Oak Ridge, Tennessee.

Rupp, E. M., 1980, "Age Dependent Values of Dietary Intake for Assessing Human Exposures to Environmental Pollutants," *Health Physics*, 39, pp. 151-163.

Rusch, G. M., 1993, "The History and Development of Emergency Response Planning Guidelines," *Journal of Hazardous Materials*, 33, pp 193-202, Elsevier Publishers, Amsterdam.

Sanford, G. B., 1993, *Specific Manufacturing Capability Project Hazards Assessment*, BWI-1333, Babcock & Wilcox Company, Idaho Falls, Idaho, March.

Schindler, R. E., 1991, letter to P. B. Summers, regarding "Distribution of Airborne Emissions Evaluation for the HLWTFR Project," Schi-08-91, Westinghouse Idaho Nuclear Company, Inc., Idaho Falls, Idaho, May 16.

Schwendiman, L. C., 1977, *Supporting Information for the Estimation of Plutonium Oxide Leak Rates through Very Small Apertures*, BNWL-2198, Battelle Pacific Northwest Laboratories, Richland, Washington, January.

Slade, D. H. (Ed.), 1968, *Meteorology and Atomic Energy*, AEC/TID-24190, U.S. Atomic Energy Commission, Silver Springs, Maryland.

Start, G. E., National Oceanic and Atmospheric Administration, Idaho Falls, Idaho, 1989, Letter to L. Finch, Westinghouse Idaho Nuclear Corporation, Inc., regarding "Interim guidance for dispersion calculations at ICPP," March 24, 1989.

Tkachyk, J. W., EG&G Idaho Inc., 1992, PROFS note to J. E. Medema, U.S. Department of Energy-Idaho Operations Office, regarding "TAN-734-001 unabated EDE from filter measurements," Idaho National Engineering Laboratory, Idaho Falls, Idaho, May 28.

TOXnet, 1993, *Toxicology Data Network*, National Library of Medicine, U.S. Department of Health and Human Services, Bethesda, Maryland.

URS (URS/John A. Blume & Associates), 1991, *Seismic Analysis and Evaluation of Waste Tank Vaults WM-180 and WM-187 through WM-190 at the Idaho Chemical Processing Plant*, San Francisco, California, October.

Weitzman, D. J. (EH-412), 1992, U.S. Department of Energy memorandum to J. E. Fairobent (OE-11); regarding "Status of Emergency Response Planning Guides (ERPGs)," Washington, D.C., September 23.

Wenzel, D. R., 1993, *The Radiological Safety Analysis Computer Program (RSAC-5)*, WINCO-1123, Westinghouse Idaho Nuclear Company, Inc., Idaho Falls, Idaho.

WINCO (Westinghouse Idaho Nuclear Company, Inc.), 1990, *Idaho Chemical Processing Plant (ICPP) Plant Safety Document*, "Aqueous Liquid Waste Management," WIN 107-4.2, Rev. 4a, Idaho Falls, Idaho, June.

APPENDIX A

ACCIDENT SCREENING METHODOLOGY

CONTENTS

TABLES

APPENDIX A

ACCIDENT SCREENING METHODOLOGY

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This appendix describes the accident screening methodology used in the accident assessment for the *Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Environmental Impact Statement* (SNF & INEL EIS). It screens and selects accident events (Section A-1), accident locations and material quantities involved in the events (Section A-2), and types of initiating events (Section A-3). Accident frequency ranges are estimated in Section A-4. Section A-5 summarizes and categorizes the selected accident events. Section A-6 screens multiple facility accidents in response to Defense Nuclear Facilities Safety Board concerns, and references are listed in Section A-7.

A-1 Screening and Selection Process

This section describes the process used to screen the many events that could lead to accidental release of radioactive or hazardous material or both. The screening identifies a class of bounding or most severe events that would encompass the consequences of other events with respect to the consequences away from the accident site or immediate local area. Potential consequences to the following entities are used to screen the events:

- **Member of the public at the nearest site boundary.** A hypothetical resident at the nearest site boundary relative to the location where the facility accident release has occurred

- **Worker.** A worker located 200 meters downwind of the location where the facility accident release has occurred

- **Environment.** The area outward from 100 meters downwind of the location where the facility accident release has occurred.

The screening and selection process focused on identifying those postulated events with the potential to cause consequences to a member of the public at the nearest site boundary locations. Thus this screening process may not identify bounding consequences to the facility worker within the facility or within 100 meters of the facility accident location. These consequences to the worker are addressed in part by analyzing potential accident consequences in terms of worker injuries, deaths, or exposures

Table A-1. Locations with sufficient quantities of radioactive or hazardous material to cause consequences to a member of the public under accident conditions.

Spent nuclear fuel, waste, and activity types	Idaho National Engineering Laboratory locations[a]								
	ICPP	ANL-W	TRA	TAN	RWMC	CFA	ARA/PBF	IRC	NRF
Spent nuclear fuel	Yes	Yes	Yes	Yes	No	No	Yes	No	Yes
High-level waste	Yes	No	No	No	No	No	No	No	No
Transuranic waste	No	No	No	No	Yes	No	No	No	No
Low-level waste	No	No	No	No	Yes	No	Yes	No	No
Mixed low-level waste	No	Yes	No	No	Yes	No	Yes	No	No
Hazardous waste and toxic material	Yes	Yes	No	Yes	No	Yes	Yes	Yes	No
Decontamination and decommissioning	Yes	Yes	Yes	No	No	No	Yes	No	No
Remediation	No	No	No	No	Yes	No	No	No	No

a. Location acronyms:
ANL-W - Argonne National Laboratory-West
ARA - Auxiliary Reactor Area
CFA - Central Facilities Area
ICPP - Idaho Chemical Processing Plant
IRC - INEL Research Center
NRF - Naval Reactors Facility
PBF - Power Burst Facility
RWMC - Radioactive Waste Management Complex
TAN - Test Area North
TRA - Test Reactor Area

environment, or public at the nearest site boundary locations. This accident analysis focuses on those initiating event types with sufficient energy and magnitude to cause consequences in accordance with previously established definitions for a member of the public at the nearest site boundary locations.

Initiating events were defined in three broad categories as follows:

- *External initiators* originate outside the facility and may impact the ability of the facility to maintain confinement of radioactive or hazardous material. These initiators may be related to fires and explosions nearby or caused by events at co-located facilities.

- *Internal initiators* originate within a facility and are a result of facility operations (for example, equipment failures or human error).

- *Natural phenomena initiators* include weather-related, volcanic, and seismic events.

All types of initiators were defined in terms of those events that cause or may lead to a release of materials by failure or bypass of confinement.

Weather-related events at the INEL that have sufficient energy potential to cause a release are limited to high winds and floods. Tornadoes of sufficient size are less likely than high winds. Seismic events of sufficient magnitude may occur. Volcanic events have occurred as recently as 2,100 years ago in the INEL area (Craters of the Moon National Monument). Range fires may reach facilities if accompanied by high wind. Co-located facilities may cause events such as fires, explosions, or others that may impact a facility's ability to maintain confinement of materials.

Internal initiator types vary depending upon the type of facility and operation. In most facilities, major activities are various material-handling activities. Material-handling events may cause mechanical or physical damage that could result in a release. Most facilities are designed with confinement ventilation systems with filters that prevent the release of materials to the environment. These filter systems could fail and result in a release. Confinement failures could be caused by internal fires or explosions and other events that impair or fail the ability to maintain confinement. Where sufficient quantities of fissile material exist, an inadvertent criticality could occur, which might cause a release of fission products.

Each INEL facility area was screened for initiating events with the potential to cause nonnegligible consequences. Only those locations identified in Table A-1 as having sufficient quantities of materials were considered. The results of the initiating event screening are summarized in Tables A-2 through A-9 for spent nuclear fuels, five waste types, and two types of environmental restoration activities.

Table A-4. Transuranic waste accidents: Screening results for reasonably foreseeable accident initiators with potential for releases that could cause consequences to the worker, environment, or a member of the public at the nearest site boundary.

Accident event types	Location at Idaho National Engineering Laboratory Radioactive Waste Management Complex
External Initiators	
Seismic	Yes
Aircraft impact	Yes
High winds	No
Flood	Yes[a]
Co-located facility events	No
Volcanic lava flow	Yes
External fire/explosion	Yes
Internal Initiators	
Criticality	Yes
Container-handling accident	No
Confinement failure	No
Fire/explosion	Yes
Stack/vent release	Yes

a. Floods at the Radioactive Waste Management Complex from local runoff have occurred in the past but have not resulted in offsite impacts.

Table A-5. Low-level waste accidents: Screening results for reasonably foreseeable accident initiators with potential for releases that could cause consequences to the worker, environment, or a member of the public at the nearest site boundary.

	Location at Idaho National Engineering Laboratory		
Accident event types	Argonne National Laboratory-West	Radioactive Waste Management Complex	Auxiliary Reactor Area/Power Burst Facility
External Initiators			
Seismic	No	Yes	Yes
Aircraft impact	Yes	Yes	Yes
High winds	No	No	No
Flood	No	No	No
Co-located facility events	No	No	No
Volcanism	Yes	Yes	No
External fire/explosion	No	Yes	No
Internal Initiators			
Container-handling	No	No	No
Confinement failure	No	No	No
Fire/explosion	No	Yes	Yes
Stack release	No	Yes	Yes

Table A-8. Decontamination and decommissioning accidents: Screening results for reasonably foreseeable accident initiators with potential for releases that could cause consequences to the worker, environment, or a member of the public at the nearest site boundary.

	Location at Idaho National Engineering Laboratory			
Accident event types	Idaho Chemical Processing Plant	Argonne National Laboratory-West	Test Reactor Area	Auxiliary Reactor Area/Power Burst Facility
	External Initiators			
Seismic	No	No	No	
Aircraft impact	Yes	Yes	Yes	Yes
High winds	No	No	No	No
Flood	No	No	No	No
Co-located facility	No	No	No	No
Volcanism	No	No	No	No
External fire/explosion	No	No	No	No
	Internal Initiators			
Criticality	No	No	No	No
Transportation and/or	Yes	Yes	Yes	Yes
Confinement failure	No	No	No	No
Fire/explosion	Yes	No	No	No
Stack/vent release	Yes	No	Yes	Yes

Table A-9. Remediation accidents: Screening results for reasonably foreseeable accident initiators with potential for releases that could cause consequences to the worker, environment, or a member of the public at the nearest site boundary.

	Location at Idaho National Engineering Laboratory
Accident event types	Radioactive Waste Management Complex
	External Initiators
Seismic	Yes
Aircraft impact	Yes
High winds	No
Flood	No
Co-located facility events	No
Volcanism	Yes
External fire/explosion	Yes
	Internal Initiators
Container-handling accident	Yes
Confinement failure	No
Fire/explosion	Yes
Stack/vent release	Yes

Table A-11. High-level waste accidents: Annual frequency ranges for initiator types with potential for releases that could cause consequences to the worker, environment, or a member of the public at the nearest site boundary.

	Location at Idaho Chemical Processing Plant			
Accident event types	High-level waste tanks	Calcine bins	New Waste Calcining Facility	Atmospheric Protection System
External Initiators				
Seismic (with subsequent fires)	10^{-3} to 10^{-4}	10^{-4} to 10^{-5}	10^{-5} to 10^{-6}	10^{-3} to 10^{-4}
Aircraft impact	10^{-6} to 10^{-7}	10^{-6} to 10^{-7}	10^{-6} to 10^{-7}	10^{-6} to 10^{-7}
Internal Initiators				
Criticality	10^{-5} to 10^{-6}	Not applicable	Not applicable	Not applicable
Fire/explosion	10^{-5} to 10^{-6}	Not applicable	10^{-5} to 10^{-6}	10^{-4} to 10^{-5}
Stack/vent release	Not applicable	Not applicable	10^{-3} to 10^{-5}	10^{-4} to 10^{-5}

Table A-12. Transuranic waste accidents: Annual frequency ranges for initiator types with potential for releases that could cause consequences to the worker, environment, or a member of the public at the nearest site boundary.

	Location at Idaho National Engineering Laboratory
Accident event types	Radioactive Waste Management Complex
External Initiators	
Seismic	10^{-3} to 10^{-4}
Aircraft impact	10^{-6} to 10^{-7}
Volcanism	10^{-4} to 10^{-5}
External fire/explosion	10^{-6} to 10^{-7}
Internal Initiators	
Criticality	10^{-6} to 10^{-7}
Fire/explosion	10^{-3} to 10^{-6}
Stack/vent release	10^{-4} to 10^{-5}

Table A-15. Hazardous waste accidents: Annual frequency ranges for initiator types with potential for releases that could cause consequences to the worker, environment, or a member of the public at the nearest site boundary.

	Location at Idaho National Engineering Laboratory				
Accident event types	Idaho Chemical Processing Plant	Argonne National Laboratory-West	Test Area North	Central Facilities Area	INEL Research Center
External Initiators					
Seismic	No	No	No	No	No
Aircraft impact	10^{-6} to 10^{-7}	10^{-6} to 10^{-7}	10^{-6} to 10^{-7}	10^{-6} to 10^{-7}	10^{-6} to 10^{-7}
Volcanism	No	No	No	No	No
External fire/explosion	No	No	No	No	No
Internal Initiators					
Movement/handling accident	10^{-2} to 10^{-3}	10^{-4} to 10^{-5}	10^{-2} to 10^{-3}	10^{-4} to 10^{-5}	10^{-5} to 10^{-6}
Fire/explosion	Not applicable	Not applicable	10^{-6} to 10^{-7}	Not applicable	10^{-5} to 10^{-6}

Table A-16. Decontamination and decommissioning accidents: Annual frequency ranges for initiator types with potential for releases that could cause consequences to the worker, environment, or a member of the public at the nearest site boundary.

	Location at Idaho National Engineering Laboratory			
Accident event types	Idaho Chemical Processing Plant	Argonne National Laboratory-West	Test Reactor Area	Auxiliary Reactor Area/Power Burst Facility
External Initiators				
Aircraft impact	10^{-6} to 10^{-7}	10^{-6} to 10^{-7}	10^{-6} to 10^{-7}	10^{-6} to 10^{-7}
Internal Initiators				
Movement/handling accident	10^{-3} to 10^{-4}	10^{-3} to 10^{-4}	10^{-3} to 10^{-4}	10^{-3} to 10^{-4}
Fire/explosion	10^{-3} to 10^{-4}	Not applicable	Not applicable	Not applicable
Stack/vent release	10^{-3} to 10^{-4}	10^{-3} to 10^{-4}	10^{-3} to 10^{-4}	10^{-3} to 10^{-4}

Volcanism is prevalent throughout the site area and has occurred within the last several thousand years at the Craters of the Moon National Monument to the southwest of the site. The Radioactive Waste Management Complex is the nearest location to the most recent Craters of the Moon lava flow. Frequency estimates have been extrapolated from a potential flow at Craters of the Moon to the potential for a lava flow engulfing the Radioactive Waste Management Complex. The frequency of this event is estimated to be 10^{-4} to 10^{-5} per year.

Frequency of high winds at the site is derived from historical weather records and extrapolated to a natural phenomena occurrence report for DOE sites (Kennedy et al. 1990). In general all permanent construction is designed to withstand very infrequent high winds of more than 80 miles per hour with no damage. Wind speeds much greater than 80 miles per hour would be required to cause damage to any permanent structure at the site; furthermore the expected frequency of these wind speeds is very small. Some remediation projects, however, may be conducted outdoors using temporary buildings and thus may be more susceptible to high winds. A frequency range of 10^{-5} to 10^{-6} per year has been estimated for these type of remediation accidents due to high winds.

The external fire/explosion event of interest is a large propane tank accident at the Radioactive Waste Management Complex. Currently, this tank is located close to storage and operating areas but will be moved farther from the areas to preclude any site operation from damaging the tank and causing an accident. However, the tank is large enough to cause a large explosion under certain conditions. The frequency of this accident is estimated at 10^{-6} to 10^{-7} per year (EG&G Idaho 1986).

A-4.2 Estimates of Spent Nuclear Fuel Internal Event Annual Frequencies

Four internal events that could occur at spent nuclear fuel locations at the INEL include inadvertent criticalities, fuel handling mechanical damage, cask failure, or fuel pool drainage.

Inadvertent criticalities could occur wherever sufficient fissile material such as spent nuclear fuel is located. Historically, no criticalities have occurred during spent nuclear fuel operations and storage. Criticality conditions are well known and very well protected against with multiple independent protection failures required for an event to occur. Frequencies for criticalities are developed for some operations but are not generally referenced and are in the range of 10^{-3} to 10^{-5} depending upon the magnitude of operations at a given location (Vail 1993).

Fuel handling mechanical damage scenarios may result in releases. These events are estimated to be in the 10^{-2} to 10^{-3} range.

Cask failures could occur during any operation involving spent nuclear fuel. Casks are designed to withstand large impacts and dropping events and are of extremely rugged construction. The type of event that could cause a cask to fail and a subsequent release would be of low frequency, in the 10^{-5} to 10^{-6} range.

A-5 Accident Event Selection and Categorization Summary

The selected accident events are categorized in Table A-18. Each waste or material type will then have events grouped by frequency interval. Then a representative or bounding event for that group can be selected for consequence assessment to represent expected consequences of the group of events for that waste type frequency interval.

A-6 Screening for Multifacility Accidents

Concern was expressed by the Defense Nuclear Facilities Safety Board that a seismically initiated accident could involve more than one facility in a given facility area and the spent nuclear fuel accidents as analyzed in the SNF and INEL EIS may not represent a bounding reasonably foreseeable scenario because (a) they involved only a small fraction of the spent nuclear fuel inventory at a facility area, and (b) a seismic event large enough to fail one facility may have implications on other facilities within a facility area. Further, the Defense Nuclear Facilities Safety Board expressed an opinion that the potential for fires following a seismic event is near-certain.

The accident assessment considered a number of factors in the screening of maximum reasonably foreseeable accidents. These included comparisons of inventory, potential initiators, source terms, release mechanisms, accident likelihood, and risk dominance. Consistent with DOE guidance, accident scenarios less likely than once in ten million years ($<10^{-7}$/yr) were considered beyond reasonably foreseeable. For accidents that could be caused by more than one initiator, the more likely initiator was analyzed because it would involve the highest risk (consequence times probability of occurrence). As stated in EIS Section 5.14.2.1, Accident Screening and Selection Process, the screening process also found that only one initiator, a large seismic event, could involve multiple facilities.

The accident analyzed for the SNF and INEL EIS with the greatest radiological consequence involves a seismic event causing fuel melting at the Hot Fuel Examination Facility in the Argonne National Laboratory-West facility area. The accident with the greatest nonradiological consequence is a release of chlorine gas at the Argonne National Laboratory-West facility area. In the sections that follow, potential for multifacility releases greater than those already considered are assessed for each facility area.

A-6.1 Idaho Chemical Processing Plant Facility Area

Spent nuclear fuel is stored in three primary areas in the Idaho Chemical Processing Plant facility area. Ranked by the inventory of spent nuclear fuel from highest to lowest, these areas are:

1. Dry storage in a number of underground, steel-lined, concrete silos at CPP-749 (largely uranium-238) that are not seismically fragile or subject to forming a critical configuration easily.

2. Wet storage in a water pools in CPP-666. This is a modern facility with a filtered ventilation system and designed to current structural standards.

3. Combined wet storage in a water pools and dry storage in CPP-603. Efforts are under way to remove fuels from the CPP-603 pools because that portion of the building does not meet current structural standards, has no ventilation system, and houses vulnerable fuels as assessed in the November 1993 Spent Fuel Working Group Report.

For groundwater consequences, the failure of a high-level waste tank as analyzed in the Section 4.1.2.4 is bounding for the following reasons:

1. A completely full tank was assumed (300,000 gallons), and the composition of the waste used for the source term in the analyzed accident was assumed to have the highest curie content expected in spent nuclear fuel processing first-cycle extraction liquids.

2. Currently, tanks that formerly contained waste from spent fuel processing are empty down to heel level (15,000 gallons). Other types of waste expected to be stored in the tank farm (sodium-bearing and decontamination wastes) over the duration analyzed in the SNF and INEL EIS have a lower curie content (approximately ten times lower) than the source term used in the accident analysis. Except in Alternative D, no more spent fuel processing wastes would be generated. Under Alternative D, new tanks would be constructed to receive processing wastes.

Nonradiological hazardous or toxic materials are also present at Idaho Chemical Processing Plant. A release of chlorine is reasonably foreseeable (10^{-5}/yr) as a result of a seismic event. This accident is analyzed in Section 7.1.2.4, and could cause significant health effects to workers at the Idaho Chemical Processing Plant, but would not cause effects at or beyond the INEL boundary. The consequences are bounded by a chlorine release at Argonne National Laboratory-West.

A-6.2 Test Area North Facility Area

The Test Area North facility area contains spent nuclear fuel storage in the storage pool in TAN-607 Hot Cell Complex and in dry storage casks on the TAN-791 testing pad west of TAN-607. Fuel may be present in the TAN-607 Hot Shop for limited periods of time during fuel handling operations. Four casks are used to test dry storage technologies at the TAN-791 test pad. Criticality accidents have been analyzed in the Test Area North facility safety analysis reports (EG&G Idaho, 1991; Franz, 1991). A criticality during dry storage has been analyzed in the facility safety analysis report and is not reasonably foreseeable. A criticality caused by a seismic event during handling of dry storage casks was analyzed in Section 3.1.3.2, and was determined to have a frequency in the range of 10^{-6} to 10^{-7} per year. More than one criticality accident at Test Area North would require a criticality event in the storage pool and would also require that casks were concurrently being handled in the Hot Shop with a coincident criticality accident. Coincident criticality events are not considered reasonably foreseeable.

Fires have also been analyzed in the Test Area North facility safety analysis reports. Fires resulting in a radioactive release from the spent nuclear fuel stored at Test Area North are beyond reasonably foreseeable because combustible materials are kept at a minimum in the vicinity of the storage locations. Most combustible materials are those associated with electrical cables and controls. If a fire were to occur, release of radioactive material to the environment would not be expected unless the fire started in or spread to the TAN-607 Hot Cell filters. The consequences of a filter fire accident at Test Area North are bounded by the filter fire accident at the Idaho Chemical Processing Plant analyzed in report indicate that such an event would not pose a health risk to facility workers or the public because of the concrete structure backed by sand and earth fill and building filtration. A coincident fire in pyrophoric fuel at the ZPPR facility would have a negligible effect on the Argonne National Laboratory-West facility seismic accident release. A fire resulting in a release from other spent nuclear fuel storage areas in this facility area is not reasonably foreseeable because there are limited combustibles in the vicinity of the fuels and the fuels are not pyrophoric. The analyzed Hot Fuel Examination Facility and Fuel Cycle Facility Hot Cell seismically initiated accidents used conservative assumptions for failure of multiple barriers, material at risk, source term, and atmospheric conditions. In addition, the accidents assumed fresh fuel from the Experimental Breeder Reactor II reactor (now shut down) was involved. Coincident events at the Hot Fuel Examination Facility and Fuel Cycle Facility that exceed the consequences of the analyzed single facility accidents are not reasonably foreseeable.

Comparison of spent nuclear fuel storage facilities at Argonne National Laboratory-West for seismic initiators

Parameter	Hot Fuel Examination Facility	Radioactive Scrap and Waste Facility	Zero Power Physics Reactor	Transient Reactor Test facility
Seismic criticality likelihood	<1E-07/yr	<1E-07/yr	<1E-07/yr	<1E-07/yr
Release fraction	Retention	Retention	Retention	Retention
Fire impact to release	Minimal combustibles	Minimal combustibles	Negligible	Minimal combustibles

Nonradiological hazardous or toxic materials are also present at Argonne National Laboratory-West. A release of chlorine is reasonably foreseeable (10^{-5}/yr) as a result of a seismic event. This accident is the bounding seismically initiated nonradiological accident analyzed in the accident assessment (Section 7.1.2.1), and could result in significant health effects to workers in the facility area, but would not cause effects at or beyond the INEL boundary.

A-6.4 Test Reactor Facility Area

The Test Reactor Area contains spent nuclear fuel storage in pools at the Advanced Test Reactor, Materials Test Reactor, and the Advanced Reactivity Measurement Facility/Coupled Fast Reactivity Measurement Facility (ARMF/CFRMF). As at the Idaho Chemical Processing Plant pool facilities, criticalities are reasonably foreseeable (10^{-3} to 10^{-4}/yr) at these pools as a result of fuel handling errors. Earthquakes are considered in the ARMF/CFRMF facility safety analysis report, and a release is not credible ($<10^{-7}$/yr) because of the control elements in the reactor cores. The ARMF/CFRMF fuel also has a low preexisting fission product inventory because the reactors were operated at low or zero power. The Materials Test Reactor is an older facility, and a seismically initiated criticality is considered credible

Processing Plant and subsequently make these shipments. ECF also would handle test specimens which were irradiated in reactor facilities at the Test Reactor Area on the INEL site. ECF would prepare these irradiated test specimens for return to the Test Reactor Area for further irradiation or for transport to other laboratories for detailed examinations.

By the time the Record of Decision on the SNF and INEL EIS is issued (June 1995), all reactor plants that previously operated at the Naval Reactors Facility will be shut down. The S1W reactor plant (prototype of the submarine NAUTILUS) and the A1W reactor plant (prototype of the aircraft carrier ENTERPRISE) have already been shut down permanently. The S5G reactor plant (prototype of the submarine NARWHAL) will be shut down permanently in June 1995.

In view of the reactor shutdowns, the only facilities at the Naval Reactors Facility where large amounts of radioactivity might be released in the event of a severe seismic event are at ECF including the water pools and hot cells (where spent nuclear fuel is examined) and the railroad sidings at ECF where M-130s or M-140s (or equivalent dry storage containers) are being held prior to unloading into ECF water pools.

The highest risk accident at ECF is complete draining of the water pool which is initiated by a design basis seismic event (ground acceleration at the Naval Reactors Facility of 0.24g at bedrock). The probability of such an event is 10^{-5} per year, which takes into account both the likelihood of the seismic event and the likelihood that pool drainage would result. The risk to the general public within 80 kilometers (50 miles) of the INEL site for this accident is calculated to be 1.7×10^{-7} latent cancer fatalities per year among the 115,000 people in the area. All other accidents evaluated for spent nuclear fuel management at the Naval Reactors Facility are lower risk.

Multiple accidents, such as a criticality event coupled with a water pool drainage, would have a probability of occurrence of less than 10^{-7} per year and thus are classified as beyond reasonably foreseeable. Calculations have shown that major structural damage is not expected to either the fuel modules, the ECF water pools, the storage racks, or to the ECF building structure from a design basis earthquake. Structural calculations have also been made for a beyond design basis earthquake (0.40g at bedrock) and these show similar structural integrity, that is, no major structural damage. This beyond design basis earthquake is ten times less likely than the design basis event.

A severe seismic event is not expected to cause any significant releases from M-130 or M-140 shipping containers, on railroad sidings or on concrete pads near ECF (or in equivalent dry storage containers). These containers, and the navy fuel inside them, have been designed to withstand much greater forces than an earthquake.

A severe seismic event might initiate a fire at ECF, perhaps in electrical equipment. There is little combustible material at ECF. However, although unlikely, a seismic casualty could result in a high efficiency particulate air filter fire, which has a risk of latent cancer fatalities to the general

A-7 References

EG&G Idaho 1993, *Waste Experimental Reduction Facility Safety Analysis Report (Draft)*, WM-F1-81-026, Rev. 8, EG&G Idaho, Inc., Idaho Falls, Idaho, June.

EG&G Idaho, 1992, *Pit 9 Comprehensive Demonstration Preliminary Safety Analysis Report (Draft)*, EGG-WM-10186, EG&G Idaho, Inc., Idaho Falls, Idaho, May.

EG&G Idaho, 1991, *Advanced Test Reactor Probabilistic Risk Assessment*, EGG-PRP-8823, Rev. 1, EG&G Idaho, Inc., Idaho Falls, Idaho, September.

EG&G Idaho, 1986, *Safety Analysis for the Radioactive Waste Management Complex at the Idaho National Engineering Laboratory*, WM-PD-86-011, Rev. 2, EG&G Idaho, Inc., Idaho Falls, Idaho, July.

Franz, W. A., 1991, *Safety Analysis Report for Spent Fuel Storage Cask Testing Pad*, DRR-WMTO-3848, Rev. 4, EG&G Idaho, Inc., Idaho Falls, Idaho, August 20.

Lee, L. G., J. M. Mines, B. B. Webb, 1993, *Assessment of Aircraft Impact Possibilities at the Idaho Chemical Processing Plant on the INEL Site*, WINCO-1137, Westinghouse Idaho Nuclear Company, Inc., Idaho Falls, Idaho, August.

Kennedy, R. P., S. A. Short, J. R. McDonald, M. W. McCann, Jr., R. C. Murray, and J. R. Hill, 1990, *Design and Evaluation Guidelines for U.S. Department of Energy Facilities Subjected to Natural Phenomena Hazards*, UCRL-15910, Lawrence Livermore National Laboratory, Berkeley, California, June.

Vail, J. A., 1993, Westinghouse Idaho Nuclear Company, Inc., letter to D. C. Slaughterbeck, Science Applications Internations Corporation, regarding "Task 1 Description for WINCO Stored Nuclear Fuels - Existing Facilities," JAV-12-93, Idaho Falls, Idaho, December 8.

WCC (Woodward-Clyde Consultants), 1990, *Earthquake Strong Ground Motion Estimates for the Idaho National Engineering Laboratory, 3 vol.*, EGG-BG-9350, EG&G Idaho, Inc., Idaho Falls, Idaho, November.

WCC (Woodward-Clyde Consultants), 1992, *Earthquake Ground Motion Evaluations for the Proposed New Production Reactor at the Idaho National Engineering Laboratory*, 2 volumes, EGG-GEO-10304, EG&G Idaho, Inc., Idaho Falls, Idaho, June.

APPENDIX B

RADIOLOGICAL COMPARISON STANDARDS

APPENDIX B

RADIOLOGICAL COMPARISON STANDARDS

The U.S. Department of Energy's (DOE's) National Safety Policy Goal (DOE 1991) has been used to compare calculated doses and potential health effects. This Policy Goal states (in part) that the cancer fatality risk to the population in the vicinity of a DOE nuclear facility should not exceed one tenth of one percent of the sum of all cancer fatality risks resulting from all other causes. The National Safety Policy Goal represents an integrated operational and accident aiming point for DOE facilities. As such, the goal is not intended to represent an acceptance criterion. Acceptance criteria for facilities are embedded in the compliance with DOE orders and standards.

To quantify the National Safety Policy Goal, the following relationships apply. The average risk of fatal cancer to an individual (without accident) is approximately 2 chances in 1000 per year (NRC 1990). That is, in a population of 1,000 people, two fatal cancers would be expected per year. The likelihood of a fatal cancer for an individual is then 0.002 per year. The Policy Goal indicates that the risk should not be increased by more than one tenth of one percent (0.1 percent or a fraction 0.001). Multiplying the average likelihood without an accident (0.002 per year) times the incremental increase (0.001) suggested as a goal by DOE results in an annual risk of cancer fatalities from DOE facilities of 0.000002, or 2×10^{-6} per year. This value can be compared to the likelihood that a hypothetical person at the nearest site boundary would have a fatal cancer from radiological accidents.

References

NRC (National Research Council), 1990, *Health Effects of Exposure to Low Levels of Ionizing Radiation*, Committee on the Biological Effects of Ionizing Radiation, National Research Council, Washington, D.C.

DOE (U.S. Department of Energy), 1991, Secretary of Energy Notice, "Nuclear Safety Policy," SEN-35-91, U.S. Department of Energy, Washington, D.C., September 9.

APPENDIX C

QUALITATIVE ANALYSIS OF INVOLVED WORKER CONSEQUENCES

CONTENTS

APPENDIX C

QUALITATIVE ANALYSIS OF INVOLVED WORKER CONSEQUENCES

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C-1 Methodology and Screening Evaluations

Analyses to evaluate the consequences of numerous accident scenarios for persons at 100 meters (328 meters), nearest plant access, nearest site boundary, and various towns in the vicinity of the Idaho National Engineering Laboratory are described in Chapters 3 through 8. To qualitatively assess the effects of these accidents on involved or close-in workers (less than 100 meters), the accident scenarios were screened to obtain a representative set of maximum reasonably foreseeable accidents for involved workers. The representative accidents were then subjected to further analysis. The accident scenarios for the close-in worker fall into three initiating categories: (a) equipment failure and human error events, (b) criticality events, and (c) external events. Equipment failures and human errors include events such as fuel handling accidents, inadvertent cutting into fuel, crane failures, vehicle accidents, glove box failures, shielding removal, and personnel contamination; external events include severe earthquakes, floods, airplane crashes, wind-generated missiles, or other severe weather-related phenomena.

The qualitative analysis methodology for the involved worker consisted of a screening review to select representative accident scenarios. The first screening task quantified the worker population within 100 meters of the accident site if the accident were outside or in the enclosed volume associated with an accident if the accident were within a building. The second screening task characterized the accident with respect to existence of plume rise. Plume rise effects can carry all or part of the airborne material above the involved worker and tend to lessen the accident's effect on the involved worker. The screening methodology also evaluated evacuation feasibility. Under certain accident situations, evacuation impediments may exist because of building failure or low visibility due to fire. Identification of shielding that would reduce worker exposure was also a criterion for determining a representative set of accidents. The last item of the screening methodology gave a qualitative evaluation of exposure potential.

The accident scenarios were organized into the following categories:

- Spent nuclear fuel
- High-level waste
- Transuranic waste
- Low-level waste
- Mixed low-level waste

Table C-1-2. Summary of screening criteria for high-level waste accidents.

Accident	Exposed population	Plume rise	Evacuation impediment	Shielded	Exposure potential
New Waste Calcining Facility (NWCF) stack release	<50	Yes	No	No	Low
Atmospheric Protection System (APS) seismic stack failure	<50	No	No	No	High
High-Level Waste (HLW) tank seismic[a]	<20	No	No	Yes	Low
Calcine bin - seismic	<20	No	No	No	Moderate
APS filter fire stack release	<50	Yes	No	Yes	Low
HLW tank criticality	<20	No	No	Yes	Low
HLW tank fire/explosion	<20	Yes	No	No	Low
NWCF seismic or fire/explosion	<20	Yes	No	No	Moderate
Calcine bin - aircraft impact	<20	Yes	No	No	Moderate

a. Major effect is long-term potential contamination of groundwater.

Table C-1-3. Summary of screening criteria for transuranic waste accidents.

Accident	Exposed population	Plume rise	Evacuation impediment	Shielded	Exposure potential
Radioactive Waste Management Complex (RWMC) Transuranic Storage Area (TSA) explosion	<20	Yes	Yes	No	High
RWMC TSA seismic	<20	No	Yes	No	High
RWMC Waste Characterization Facility vent release	<10	Yes	Yes	No	Low
RWMC lava flow (RWMC TSA fire - C)[a]	<20	Yes	No	No	Low
RWMC TSA fire[b]	<20	Yes	Yes	No	Low
RWMC aircraft impact[c]	<20	Yes	No	No	High
RWMC external fire/explosion	<20	Yes	Yes	No	High
RWMC criticality	<5	No	No	No	High

a. Alternative C (Minimum Treatment, Storage, and Disposal); see Chapter 1. Close-in personnel assumed to be only those individuals monitoring movement and effect of lava flow. Other personnel assumed to have adequate time for evacuation. Monitoring personnel assumed to have appropriate safety equipment.
b. Assume evacuation delayed due to initial fire fighting efforts.
c. Exposure to personnel not affected by impact.

Table C-1-5. Summary of screening criteria for mixed low-level waste accidents.

Accident	Exposed population	Plume rise	Evacuation impediment	Shielded	Exposure potential
Radioactive Waste Management Complex (RWMC) Transuranic Storage Area (TSA) explosion	<20	Yes	Yes	No	Moderate
RWMC TSA seismic	<20	No	Yes	No	Low
Waste Experimental Reduction Facility (WERF) seismic	<20	No	Yes	No	Low
RWMC Waste Characterization Facility (WCF) vent release	<20	Yes	No	No	Low
WERF stack release	<20	Yes	No	No	Low
WERF fire/explosion	<20	Yes	Yes	No	Moderate
RWMC lava flow	0	Yes	No	No	Low
WERF Waste Storage Building (WWSB) fire - C,D[a]	<20	Yes	No	No	Moderate
RWMC TSA fire	<20	Yes	No	No	Low
WWSB fire	<20	Yes	No	No	Low
RWMC external fire/explosion	<20	Yes	No	No	Moderate
RWMC criticality	<5	No	No	No	High
WWSB major fire	<20	Yes	No	No	Moderate

a. Alternatives C (Minimum Treatment, Storage, and Disposal) and D (Maximum Treatment, Storage, and Disposal); see Chapter 1.

Table C-1-7. Summary of screening criteria for environmental remediation and decontamination and decommissioning accidents.

Accident	Exposed population	Plume rise	Evacuation impediment	Shielded	Exposure potential
Remediation Pit 9 stack/vent release	<20	Yes	No	No	Low
Decontamination and decommissioning stack vent release	<50	Yes	No	No	Low
Decontamination and decommissioning movement/handling accident	<50	No	No	No	Low
Decontamination and decommissioning fire/explosion	<50	Yes	Yes	No	Low
Remediation Pit 9 fire/explosion	<20	Yes	Yes	No	Low
Remediation Pit 9 container handling outside	<10	No	No	No	Low
Remediation Pit 9 major fire	<20	Yes	No	No	Low
Remediation Pit 9 high winds	<20	No	No	No	Low
Remediation Pit 9 seismic	<20	No	Yes	No	Low

Table C-1-8. Representative accidents for involved worker analysis.

Accident	Category/initiator	Source	Exposed population
Test Area North criticality	Criticality/seismic	Spent nuclear fuel	33
Idaho Chemical Processing Plant stack collapse	External/seismic	High-level waste	50
Radioactive Waste Management Complex Transuranic Storage Area fire	Equipment failure and human error	Transuranic waste	20
Argonne National Laboratory-West chlorine release	Equipment failure and human error	Hazardous materials	100

RSAC-5 does not have the capability to calculate the quantity of activation products produced from fuel irradiation. Therefore, the activation product quantities (GNS 1985) calculated by ORIGEN2 (RSIC 1991) were adjusted to the 18.135 metric tons of uranium content of the 39 fuel assemblies. RSAC-5 allows for fractionation of nuclide quantities from fission but not for quantities of other radionuclides. To account for fractionation of the activation products, calculations were performed to adjust activities before entering the quantities as input to RSAC-5. Activation products would be generated by a criticality excursion, but the quantity would be insignificant in comparison to the quantities of activation products generated during reactor operation.

All of the material at risk (that is, 39 pressurized water reactor spent nuclear fuel assemblies) was assumed to be damaged in the event. The release from the fuel was assumed here to be a one-hour exponential release, which is considered conservative. Regulatory Guides 3.33 (NRC 1977), 3.34 (NRC 1979a), and 3.35 (NRC 1979b) assume an eight-hour release of fission products from a solution criticality. Regulatory Guide 1.25 (NRC 1972) provides release fractions for the analysis of damaged spent nuclear fuel assemblies. However, Regulatory Guide 1.25 does not address volatiles, halogens (other than iodine), or particulates. For this analysis, the release fractions for volatile fission products was assumed to be the same as the release fraction for noble gases. The release fraction for all halogens was assumed to be the same as the iodine release fraction. The cladding crud release was assumed to be 100 percent due to the violent mechanical damage that might occur from the seismic event. The respirable fraction was assumed to be 0.1 percent for the crud and 100 percent for the other nonvolatile solids. These release fractions are considerably lower than the release fractions recommended in Regulatory Guides 3.33, 3.34, and 3.35, but they are more appropriate for the Test Area North criticality scenario because the Regulatory Guide release fractions are for a solution/slurry criticality and there are no spent nuclear fuel solutions or slurries at TAN-607. Further, there is no potential for fuel melting due to decay heat since the Test Area North spent nuclear fuel has cooled for 10 years. Melted fuel may have release fractions that would be closer to the Regulatory Guide values.

There is no credit taken for any reduction and removal, such as plate-out, scrubbing, or filtration. Regulatory Guide 1.25 provides pool decontamination factors, but they are not considered appropriate here since the extent of fuel flooding is relatively small compared to the 23 feet assumed in Regulatory Guide 1.25.

C-2.3 Dose Calculations and Results

The involved worker was exposed to both the airborne releases of radioactive material and the prompt neutron and gamma flux from the criticality. The RSAC-5 code (Wenzel 1993) cannot be directly applied to simulate airborne dispersion and exposure to receptors at distances of less than 100 meters. As an alternative, user-defined dispersion coefficients were hand-calculated and input to RSAC-5. This retained the source term and dose calculation features of the code while simulating the activity concentrations predicted for the involved worker. Air dispersion to the involved worker was modeled by postulating an initial, uniform dispersion of the source term into a finite volume

Table C-2-2. Summary of risk calculation results for TAN-607 criticality event.

Distance (m)	Involved population	Total dose prompt (shielded) plus airborne (rem)	Risk (fatal cancers per year)
20	1	85	3.4E-06
40	4	82	9.7E-06
60	7	81	1.6E-05
80	9	81	2.3E-05
100	12	81	2.9E-05
Total	33		8.1E-05

C-3 Representative High-Level Waste Accident for Involved Worker: Idaho Chemical Processing Plant Earthquake-Induced Main Stack Collapse

The potential consequences to an involved worker as a result of a seismically induced toppling of the main stack at the Idaho Chemical Processing Plant (ICPP) caused by a large seismic event were analyzed to qualitatively assess impacts to an involved worker from high-level waste accidents. The analysis for the involved worker was a modification of the uninvolved worker analysis for the stack collapse described in Section 4.1.2.2.

C-3.1 Description of Accident

The airborne radiological release due to the toppling event resulted from crushing of filter equipment containing radionuclides. The ICPP main stack is the final release point for gaseous waste streams from several ICPP facilities. The off-gas is passed through treatment equipment--mist eliminators, condensers, prefilters, and high-efficiency particulate air (HEPA) filters--before entering the stack. The stack is 76 meters (250 feet) high with an 2.4-meter (8-foot) inside diameter tapering to 2 meters (6.5 feet) at the top and is constructed of a stainless-steel liner encased in a layer of foam, a layer of brick, and two external sheaths of reinforced concrete. The total weight of the stack is over 3 million pounds.

If the stack is assumed to topple in any direction and if the leverage of the stack base is included, the collapsed stack could impact the area within a 91-meter (300-foot) radius circle with the center at the stack base (Chung 1993). The damage analysis (Chung 1993) indicated that major damage would generally result from stack impact. This analysis identified the bounding radiological airborne release as a CPP-604 vessel off-gas HEPA filter F-WL-121 and CPP-756 Atmospheric Protection System ventilation prefilter crushing accident.

Table C-3-1. Material at risk for main stack toppling accident at the Idaho Chemical Processing Plant.

Nuclide	(A) HEPA Filter F-WL-121 at 25 R/h (Ci)	(B) Atmospheric Protection System prefilter CPP-756 future projection (Ci)	(A) + (B) Material at risk (Ci)
Strontium-90	1.18E+00	5.26E+00	6.44E+00
Yttrium-90	1.18E+00	5.26E+00	6.44E+00
Ruthenium-106	1.57E−01	2.24E+01	2.26E+01
Rhodium-106	1.57E−01	2.24E+01	2.26E+01
Antimony-125	3.92E−02	3.66E+01	3.66E+01
Cesium-134	7.85E−01	7.32E−01	1.52E+00
Barium-137m	1.18E+00	1.24E+01	1.36E+01
Cesium-137	1.18E+00	1.35E+01	1.47E+01
Uranium-234	Not applicable	2.54E+01	2.54E+01
Plutonium-239	2.75E−05	2.20E+00	2.20E+00
Americium-241	1.58E−05	1.46E−01	1.46E−01

Table C-3-2. Important contributors to source term for main stack toppling accident at Idaho Chemical Processing Plant.

Nuclide	Source term (Ci)
Strontium-90	0.0644
Yttrium-90	0.0644
Ruthenium-106	0.2260
Rhodium-106	0.2260
Antimony-125	0.3660
Cesium-134	0.0152
Barium-137M	0.1360
Cesium-137	0.1470
Uranium-234	0.2540
Plutonium-239	0.0220
Americium-241	0.0015

Table C-3-3. Summary of dose calculation results for main stack toppling event at Idaho Chemical Processing Plant (earthquake).

Distance (m)	Involved population	Dose (rem)	Risk (fatal cancers per year)
20	2	3487	1.0E-03
40	6	2605	2.3E-03
60	10	1671	2.5E-03
80	14	1143	2.4E-03
100	18	829	2.2E-03
Total	50		1.1E-02

C-4 Representative Transuranic Waste Accident for Involved Worker: Fire and Explosion at the Radioactive Waste Management Complex

This section describes a representative accident resulting in the exposure of an involved worker to a low-level waste release. The accident involves a fire and explosion at the Radioactive Waste Management Complex (RWMC).

The Transuranic Storage Area (TSA) is a 54-acre tract of land located in the southeast portion of RWMC. The primary purpose of the TSA is examination, segregation, certification, and interim storage of solid transuranic waste. The TSA consists of storage pads, the Intermediate-Level Transuranic Storage Facility, the Stored Waste Examination Pilot Plant, and support facilities. Over 100,000 waste containers are stored on these pads.

C-4.1 Description of Accident

Involved worker exposures at the RWMC are postulated to occur due to an uncontrolled fire involving stored transuranic waste in several locations. These locations were selected because the waste packages are aboveground and accessible to fire events. Two areas were identified where significant quantities of waste material could be exposed to a postulated fire:

- Transuranic waste in the TSA [Drum Vent Facility, Stored Waste Examination Pilot Plant, Air Support Weather Shield Building-II (ASWS-II), Certification and Segregation Building (C&S), or TRUPACT-II Loading Facility

- Unburied low-level waste in the Subsurface Disposal Area.

An explosion in the waste at the RWMC would mobilize some contamination, and could cause the release of potentially significant quantities of radioactivity. The places where there are at least remote possibilities of explosions involving the waste are at the TSA, the Intermediate Level

Table C-4-1. Radioactive Waste Management Complex postulated event locations, volumes, and total densities.

Location	Volume (m^3)	Density (Ci/m^3)
Transuranic Storage Area (TSA) (Air Support Weather Shield Building-II)	11,935	4.06
TSA (Drum explosion)	0.210	4.06
Intermediate-Level Transuranic Storage Facility (explosion)	0.114	38.3
Subsurface Disposal Area (low-level waste)	135,000	0.928
Waste Volume Reduction Facility (low-level waste explosion)	0.398	0.928

Table C-4-2. Isotope concentrations for Radioactive Waste Management Complex areas of interest (curies per cubic meter).

Isotope	TSA[a] (transuranic waste)	ILTSF[a] (transuranic waste)	SDA[a] (low-level waste)	WVRF[a] (low-level waste)
Uranium-233	2.03E−02	1.67E−04	(b)	(b)
Plutonium-238	9.33E−01	1.16E−01	(b)	(b)
Plutonium-239	4.00E−01	2.00E−01	(b)	(b)
Plutonium-240	9.75E−02	6.11E−02	(b)	(b)
Plutonium-241	1.40E+00	1.90E+00	(b)	(b)
Americium-241	1.21E+00	4.90E−02	(b)	(b)
Barium-137m	(b)	1.85E+01	1.54E−01	1.54E−01
Cesium-137	(b)	1.75E+01	1.63E−01	1.63E−01
Manganese-54	(b)	(b)	2.38E−05	2.38E−05
Nickel-59	(b)	(b)	9.41E−02	9.41E−02
Cobalt-60	(b)	(b)	3.76E−01	3.76E−01
Strontium-90	(b)	(b)	7.06E−02	7.06E−02
Yttrium-90	(b)	(b)	7.06E−02	7.06E−02
Total	4.06E+00	3.83E+01	9.28E−01	9.28E−01

a. TSA = Transuranic Storage Area; ILTSF = Intermediate-Level Transuranic Storage Facility; SDA = Subsurface Disposal Area; WVRF = Waste Volume Reduction Facility.
b. Not present at areas of interest.

dispersion parameters for Stability Class F meteorology, an average wind velocity of 0.5 meters per seconds, and the default INEL air density of 1.099×10^3 grams per cubic meter. Exposure at a downwind distance of 100 meters was requested, and the RSAC-calculated χ/Q at that distance was extracted from the output file. Then, a RISKIND calculation was performed using the same meteorology, requesting exposure at downwind distances of 20, 40, 60, 80 and 100 meters. The RISKIND values for χ/Q at these distances were used to scale the RSAC-calculated χ/Q at 100 meters for use in predicting exposures at the required close-in distances.

For enclosed volume scenarios, such as the transuranic waste drum explosion inside ASB-II and the Waste Volume Reduction Facility compactor explosion, a simplified χ/Q calculation was performed. This calculation assumes that the released material is uniformly distributed throughout the enclosed volume and uses the equation

$$\frac{X}{Q} = \frac{WET}{RV} \tag{C-1}$$

where χ/Q = calculated value (s/m^3)
WET = worker exposure time (s)
RV = volume in which the contaminants are distributed (m^3).

Since the model assumes a uniform distribution of contaminants, the calculated χ/Q is equally valid for inhalation and immersion exposure calculations for any distance from the release point inside the building.

The doses for both the open-air and enclosed-volume scenarios are summarized in Table C-4-4.

The doses due to the fire in the TSA are the most limiting of the events analyzed. Combining the risk conversion factor with the doses due to the TSA fire resulted in the calculated health effects shown in Table C-4-5. The results are conservative in that a general population risk conversion factor rather than a radiation worker risk conversion factor was used to convert from dose to fatal cancers per year. With an accident frequency of 2×10^{-4} per year and an involved worker population of 20, the risk of fatal cancer to an involved worker would be 7.7×10^{-4} fatalities per year. Workers near the source of the release would be at potential risk of injury or death.

Table C-4-5. Risk from fire in Radioactive Waste Management Complex Transuranic Storage Area.

Distance (m)	Involved Population	Dose (rem)	Risk (fatal cancers per year)
20	1	955	7.6E−05
40	2	711	1.7E−04
60	4	461	1.8E−04
80	6	313	1.8E−04
100	7	227	1.6E−04
Total	20		7.7E−04

C-5 Representative Hazardous Material Accident for Involved Worker: Chlorine Release at Argonne National Laboratory-West

This section describes a representative accident resulting in the exposure of an involved worker to a hazardous material release. The accident involves releasing chlorine at Argonne National Laboratory-West (ANL-W). The analysis for the involved worker is based on the uninvolved worker analysis for the ANL-W chlorine release discussed in Section 7.1.2.1.

C-5.1 Description of Accident

The total release of chlorine gas from two cylinders in a storage cabinet outside of ANL-754 was selected as being representative of maximum reasonably foreseeable events for an involved workers' exposure to hazardous materials.

The ANL-754 chlorine release could result from a number of cylinder accidents, including cylinder toppling, vehicle impact, piping rupture, or inadvertent valve opening. Any of these events could result in the total release of chlorine. Because of the large number and variety of these events, no attempt was made to identify a specific scenario. The frequency for a release from a pressure vessel is 1×10^{-4} per tank per year (FEMA/EPA/DOT 1989). A common-cause failure causing the release of two tanks was assumed to occur about 10 percent of the time. Therefore, for a postulated total release of two cylinders, the estimated likelihood is 1×10^{-5} per year.

C-5.2 Development of Toxic Chemical Source Term

A small cabinet outside of the east side of ANL-754 contains two 68-kilogram (150-pound) chlorine gas cylinders. Since the release fractions for both gases and liquids are assumed to be 1.0, the source term is equal to the material at risk.

Eimutis, E. C. and M. G. Konicek, 1972, "Derivation of Continuous Functions for the Lateral and Vertical Dispersion Coefficients," *Atmospheric Environment, 6*, pp. 859-863.

FEMA/EPA/DOT (Federal Emergency Management Agency)/U.S. Environmental Protection Agency/U.S. Department of Transportation), ca. 1989, *Handbook Of Chemical Hazard Analysis Procedures*, Washington, D.C.

GNS (General Nuclear Systems, Inc.), 1985, *Topical Safety Analysis Report For The Castor V/21 Cask Independent Spent Fuel Storage Installation (Dry Storage),"* V-21 SAR, Rev. 1, January.

Grove (Grove Engineering, Inc.), 1988, *Microshield Version 3 Manual*, Rockville, Maryland, April 12.

ICRP (International Commission on Radiological Protection), 1979, *Limits for Intakes of Radionuclides by Workers, Part 1*, ICRP Publication 30, Pergamon Press, Oxford, Great Britain.

NRC (U.S. Nuclear Regulatory Commission), 1972, *Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling Water and Pressurized Water Reactors*, Regulatory Guide 1.25, Washington, D.C., March 23.

NRC (U.S. Nuclear Regulatory Commission), 1977, *Assumptions Used for Evaluating the Potential Radiological Consequences of Accidental Nuclear Criticality in a Fuel Reprocessing Plant*, Regulatory Guide 3.33, Washington, D.C., April.

NRC (U.S. Nuclear Regulatory Commission), 1979a, *Assumptions Used for Evaluating the Potential Radiological Consequences of Accidental Nuclear Criticalities of Accidental Criticality in a Uranium Fuel Fabrication Plant,"* Regulatory Guide 3.34, Rev. 1, Washington, D.C., July.

NRC (U.S. Nuclear Regulatory Commission), 1979b, *Assumptions Used for Evaluating the Potential Radiological Consequences of Accidental Nuclear Criticalities of Accidental Criticality in a Plutonium Processing and Fuel Fabrication Plant*, Regulatory Guide 3.35, Rev. 1, Washington, D.C., July.

Pasquill, F., 1974, *Atmospheric Diffusion*, 2nd ed., John Wiley & Sons, New York, New York.

Peterson, W. B. and L. G. Laudas, 1986, *INPUFF 2.0A - Multiple Source Gaussian Puff Dispersion Algorithm User's Guide*, U.S. Environmental Protection Agency Report EPA/600/8-86/024.

RSIC (Radiation Shielding Information Center), 1991, "ORIGEN2.1, Isotope Generation and Depletion Code Matrix Exponential Method," CCC-371, RSIC Computer Code Collection, Oak Ridge, Tennessee.

Schindler, R. E., 1991, letter to P. B. Summers, regarding "Distribution of Airborne Emissions Evaluation for the HLWTFR Project," Schi-08-91, Westinghouse Idaho Nuclear Company, Inc., Idaho Falls, Idaho, May 16.